INTERNATIONAL ATOMIC ENERGY AGENCY

Conf-811046--19

THIRD TECHNICAL COMMITTEE MEETING AND WORKSHOP
ON FUSION REACTOR DESIGN AND TECHNOLOGY

Tokyo, Japan
October 5-16, 1981

622-I6-TC-392

CONF-811046--19

DE83 007643

FIRST-WALL AND BLANKET ENGINEERING DEVELOPMENT FOR MAGNETIC-FUSION REACTORS*

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*Work supported by the U. S. Department of Energy

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ABSTRACT

A number of programs in the USA concerned with materials and engineering development of the first wall and breeder blanket systems for magnetic fusion power reactors are described. Argonne National Laboratory has the lead or coordinating role, with many major elements of the research and engineering tests carried out by a number of organizations including industry and other national laboratories.

The purpose of the First Wall/Blanket/Shield (FW/B/S) Engineering Test Program is to establish an engineering data base derived from realistic models and facsimile testing. The program consists of a number of elements, with emphasis on defining the performance characteristics for the Fusion Engineering Device (FED) and a Fusion Demonstration Plant (FDP). The first wall element is concerned with thermo-mechanical testing. Tests are being conducted for normal heat flux conditions (10-100 W/cm^2) and off-normal conditions such as plasma disruptions (100-300 MW/m^2). The blanket element is concerned with studies on how to simulate fusion neutron bulk heating effects in blankets using non-nuclear techniques and fission reactors. In addition, a facility is being constructed at ANL for electromagnetic testing of FW/B/S components. The assembly/maintenance/repair (AMR) element of the program is now focusing on preparation of a designer's guide book and development of vacuum and coolant penetrations and connectors. Industrial participation in this activity includes Westinghouse Electric Corporation; General Atomic Company; EG&G Idaho, Inc.; McDonnell Douglas Astronautics Company (MDAC); and Remote Technology Corporation.

Current activities in the breeder blanket development program are focused on a number of solid breeders and include: (1) preparation and fabrication; (2) measurement of thermochemical properties and phase equilibria; (3) studies of compatibility of candidate structural materials with solid breeders; (4) determination of thermal diffusivity and thermal conductivity; and (5) small-scale neutron irradiation and tritium extraction tests for Li_2ZrO_3 and Li_2TiO_3. These experiments, in conjunction with design and modelling activities, provide support for two comprehensive irradiation tests to be performed in 1982: (A) TRIO-01, an in-pile test of tritium recovery and thermal performance of a miniaturized blanket assembly, to be conducted at Oak Ridge National Laboratory (ORNL); and (B) FUBR, a comprehensive test of irradiation performance of Li_2O, $LiAlO_2$, Li_4SiO_4, and Li_2ZrO_4, to be conducted in the Experimental Breeder Reactor-II (EBR-II) at ANL-West. Other participants include the Hanford Engineering Development Laboratory (HEDL), MDAC, ORNL, and the Japan Atomic Energy Research Institute (JAERI).

1.0 INTRODUCTION

The development of a thorough and reliable materials and engineering data base for first wall and blanket designs is a vital step in the successful achievement of demonstrating commercial fusion power. A number of programs in the USA are addressing this critical objective. This paper briefly describes two important aspects of this program, including a First Wall/Blanket/Shield Engineering Test Program and a tritium breeder materials and engineering development program.

2.0 FIRST WALL/BLANKET/SHIELD ENGINEERING TEST PROGRAM

2.1 General overview

Because the adequacy of first wall/blanket/shield (FW/B/S) concepts in the total fusion environment cannot be determined by analytical and computational methods alone, the FW/B/S Engineering Test Program (ETP) for magnetically confined fusion power reactors was established by the U. S. Department of Energy/Office of Fusion Energy (DOE/OFE) in late 1979. The overall background and scope have been described in detail in earlier reports. [1,2] ANL was designated as the lead technical organization for this program, with the prime objective of establishing, through realistic testing, an engineering data base that will accelerate progress in the proper design of FW/B/S components and systems.

To this end, a program comprised of four Test Program Elements (TPE's) has been initiated, the emphasis of which is on defining the performance parameters for the FED [3] and the FDP. These elements are:

TPE-I: Non-nuclear thermal-hydraulic and thermomechanical testing of first wall and component facsimiles with emphasis on surface heat loads and heat load transient (i.e., plasma disruption) effects.

TPE-II: Non-nuclear and nuclear testing of FW/B/S components and assemblies with emphasis on bulk (nuclear) heating effects, integrated FW/B/S hydraulics and mechanics, blanket coolant system transients, and nuclear benchmarks.

TPE-III: FW/B/S electromagnetic and eddy current effects testing, including pulsed field penetration, torque and force restraint, electromagnetic transient response, reactions of ferromagnetic materials, liquid metal MHD effects and the like.

TPE-IV: FW/B/S assembly, maintenance, and repair (AMR) studies focusing on generic AMR criteria, with the objective of preparing an AMR designer's guide book; also, development of rapid remote assembly/disassembly joint system technology, leak detection, and remote handling methods.

Contracts for the performance of TPE's I, II, and IV have been awarded to industrial organizations; TPE-III is being conducted by ANL. Near-term plans through 1985 are being developed and implemented, with long-term plans through 1990 to follow.

2.2 Test program elements

TPE-I is being conducted by the Westinghouse Electric Corporation. [4] The capabilities of the facilities for TPE-I and "test windows" are given in Fig. 1. Two first wall heat flux test stands will provide for the thermal-hydraulic and thermomechanical testing of first wall components and systems. The first, available in October 1981, will have an evacuable, 1.0 m dia. x 1.5 m long chamber capable of accommodating first wall component test pieces with surface area dimensions of approximately 1000 cm^2. A 50 kW electron beam gun with rastering capability over this area is included. Heat transfer capability is provided by a 6.9 MPa water loop which permits inlet temperatures up to 543 K. The second test stand, which will be available by March 1982 for similar but larger scale testing, will have a chamber 1.8 m dia. x 1.8 m long, with test piece capability of 1.0 m^2. A 100 kW rastering electron beam gun will be included initially, to be followed by a large area heat source. A 2 MW_t heat rejection capability is available by means of a 15.2 MPa water loop at inlet temperature up to 573 K.

Four types of test articles, presently being fabricated, are depicted in Fig. 2. The FED reference design is represented by the ribbed stainless steel flat plate. Tests to be completed by the end of 1981 include surface melting/vaporization of stainless steel and the surface recession rates, erosion rates of graphite tiles, and design margins and failure thresholds. Thermomechanical models and computer codes will also be addressed. The culmination of this work will be in validation of facilities and procedures and completion of a Detailed Technical Plan (DTP) for future testing activities.

TPE-II is being conducted by the General Atomic Company in concert with EG&G Idaho, Inc. Thermal-hydraulic/thermomechanical data requirements for blanket and shield concepts which currently appear to be feasible are being reviewed and critical testing needs identified. Also, potential FED blanket/shield test modules are being examined from the viewpoint of forming the basis for developing the non-nuclear and nuclear (later) testing strategies. The DTP for implementation of tests based on this phase of work will be completed in early 1982.

Initial emphasis will be placed on simulating the fusion bulk (nuclear) heating environment in blanket and shield facsimiles to provide test data for normal and off-normal (transient) conditions. Table I describes test articles under consideration and test conditions.

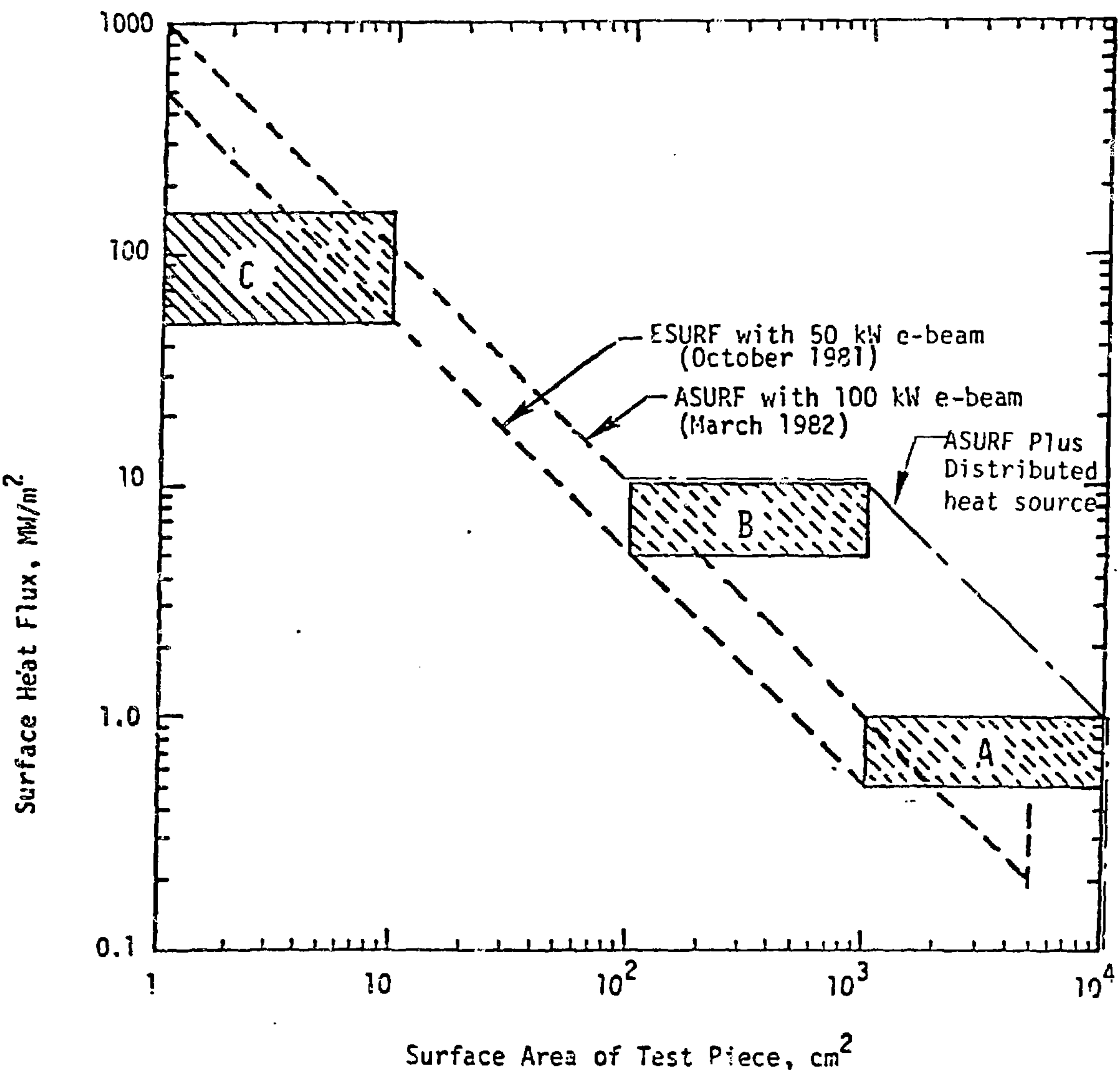


FIG. 1. COMPOSITE SURFACE HEAT FLUX OPERATING MAP FOR TPE-I.

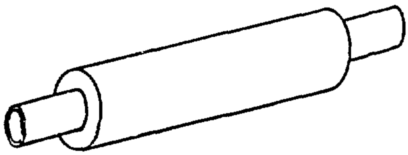
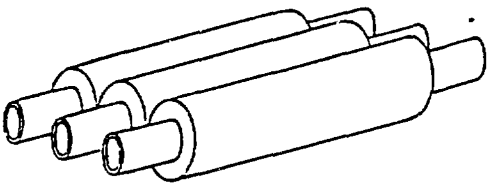
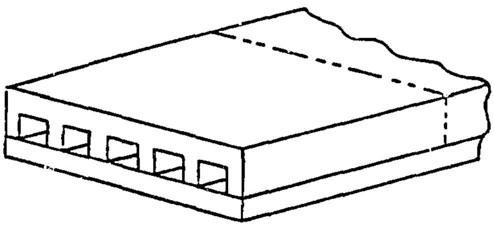
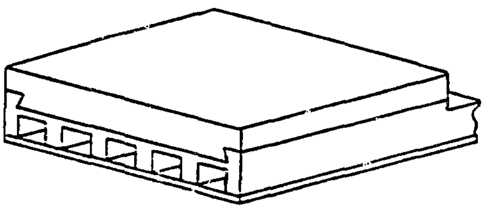
Test Piece	Description
	Single Stainless Steel Tubes
	Three Tube Facsimile of a Stainless Steel Heat Rejection Panel
	Facsimile of a Stainless Steel Flat Plate Heat Rejection Panel
	Full Scale Graphite Armor Tile, Tongue and Groove Attached to a Water-Cooled Stainless Steel Plate

FIG. 2. TEST PIECES TO BE TESTED IN PHASE O.

TABLE I

DESCRIPTION OF FW/B/S TEST ARTICLES AND TEST CONDITIONS TO BE ADDRESSED AS PART OF TPE-II

TEST OBJECTS

(a) Monolithic stainless steel blocks with integral cooling channels.

(b) Stainless steel/boron carbide composite blocks with integral cooling channels.

(c) Facsimiles of modular solid breeder canisters with integral cooling channels.

(d) Facsimiles of modular liquid metal cooled canisters with integral liquid metal flow paths.

(e) Advanced versions of articles (a) through (d) above, fitted with manifolds, connectors, and support pieces.

(f) Versions of articles (a) through (e) above, connected in series or in parallel with one or more additional articles of the same type.

TEST CONDITIONS

(1) Simulated asymmetric bulk (nuclear) heating up to 40 w/cc (peak heating rate) for normal condition testing and up to 100 w/cc for transient condition testing.

(2) Pressurized water coolant operating at coolant exit temperatures up to 573 K and coolant pressures up to 13.8 MPa. [For articles (a), (b), (c), and, where appropriate, (e) and (f) above.]

(3) Pressurized helium coolant operating at coolant exit temperatures up to 873 K and coolant pressures up to 6.9 MPa. [For articles (c) and, where appropriate, (e) and (f) above.]

(4) Liquid metal coolant operating at coolant exit temperatures up to 723 K and coolant pressures up to 3.5 MPa. [For articles (d) and, where appropriate, (e) and (f) above.]

(5) Peak structural temperatures that are consistent with recommended upper limit values based on existing materials performance data.

TPE-III is being carried out by ANL. This task addresses electromagnetic effects, both from plasma displacement or disruption and from normal or abnormal magnetic field changes. The importance of electromagnetic effects in the FW/B/S system of a fusion reactor and the uncertainty in the ability to estimate them with sufficient accuracy led to the selection of electromagnetic effects as one of the four areas for study in the FW/B/S ETP.

Important electromagnetic effects associated with the FW/B/S system are: (1) limitations on equilibrium and control field penetration to the plasma; (2) spatial perturbation of the equilibrium and control field by eddy currents; (3) forces and torques on FW/B/S components; (4) eddy current heating of FW/B/S components; (5) electrical arcing between FW/B/S components; and (6) plasma stabilization by eddy currents. These effects can arise during the following processes: (a) plasma disruption; (b) perturbations in plasma current distribution; (c) operational ohmic heating (OH) flux change; (d) operational equilibrium field (EF) change; (e) off-normal discharge of toroidal field (TF) coils; and (f) off-normal discharge of EF or OH coils.

Some of the effects which should be studied, such as the consequences of geometrical configurations like holes and segmentation, can be modelled with computer codes; others, such as assembly effects, cannot. Even the ones which can be modelled with codes must also be studied experimentally, since today's computer codes cannot treat anything but the simplest geometries and the complexities of the FW/B/S system will certainly tax the eddy current codes of the foreseeable future.

Carrying out the necessary experiments requires a facility in which values of key parameters are in a range to produce significant effects, measurable with sufficient precision to compare with theory or computations. The values must be large enough so that the results of experiments can be credibly extrapolated to a full-scale system, but small enough so that the cost of the facility is not unreasonable. The key parameters of the facility must be: (1) a sizable constant field, analagous to a tokamak toroidal field or the confining field of a mirror reactor; (2) a pulsed field with sizable rate of change, analagous to a pulsed poloidal field or to the changing field of a plasma disruption, perpendicular to the constant field; and (3) a sizable volume in order that (a) test pieces of some extent and complexity can be tested and (b) the forces, torques, currents, and field distortions which are developed are large enough to measure with adequate precision.

Analysis shows that for most experiments (measurements of field distortion, forces, and torques; model verification, and component and assembly simulations) the strength and rate of change of the pulsed field and the size of the test volume are the most important parameters; however, for experiments with saturable ferromagnetic materials, the strength of the constant (i.e., toroidal) field is the most important.

The facility being built meets these experimental needs. The constant field is modelled by a slowly pulsed solenoid field that has a rise and fall time of <3 s and a flattop of 8 s. The pulsed field is modelled by a pulsed dipole field that has a rise time of 1 s, a flattop of 3 s, and a variable decay time of 5 ms or more. The repetition rate is 1 ppm.

A horizontal solenoid field of 1.0 T is excited by six identical solenoid coils over a cylindrical volume of 0.76 m^3 (radius $r = 0.45$ m, length $\ell = 1.2$ m). Field uniformity is traded off for accessibility to the desired volume. A vertical dipole field of 0.5 T is excited by two sets of two nested dipole coils. This field is superimposed on the solenoid field; it can be forced to decay with $\dot{B}$ up to 50 T/S. Figure 3 is an isometric view of the facility. The coil parameters are given in Table II.

Two possible positions for the dipole coils were considered--inside and outside the solenoids. Inside the solenoids, the dipoles would fill a smaller volume with field and thus develop a smaller stored energy to be switched into and out of the system quickly. Also, internal dipoles would exert a smaller force density on the solenoids than external dipoles. However, external dipoles experience a smaller face density from the solenoids. As it is easier to support the solenoids against large forces than the dipoles, the decision was made to locate the dipole coils outside the solenoid.

A horizontal axis was chosen for the dipole coils, rather than a vertical axis, for ease of inserting large but delicate experiments into the test volume.

For experiments in which the overall force or torque are to be measured, the test pieces will be supported from the floor through a hole in the solenoid magnet support cylinders, rather than from the support cylinder itself. External support, similar to that in wind-tunnel experiments, protects the stress measurements from the effects of stresses which develop in the support cylinder itself during pulsing. Another advantage is that the strain gauges measuring overall force and torque can be mounted outside the magnetic field, although, of course, the strain gauges measuring local stresses must operate in the magnetic field.

Completion of the test stand and start of the experimental program is scheduled for April, 1983.

TPE-IV is a two-fold assembly, maintenance, and repair (AMR) task. A contract has been awarded to Remote Technology Corporation to survey and assess existing multi-disciplinary (e.g., nuclear, space, undersea) technology pertinent to remote AMR, which will lead to preparation of an AMR designer's guide book. A contract was also awarded to McDonnell Douglas Astronautics Corporation to pursue a related program for development of remotely operated joint systems for vacuum and coolant closures, penetrations, and joints. This will later be expanded to embrace remote AMR for a

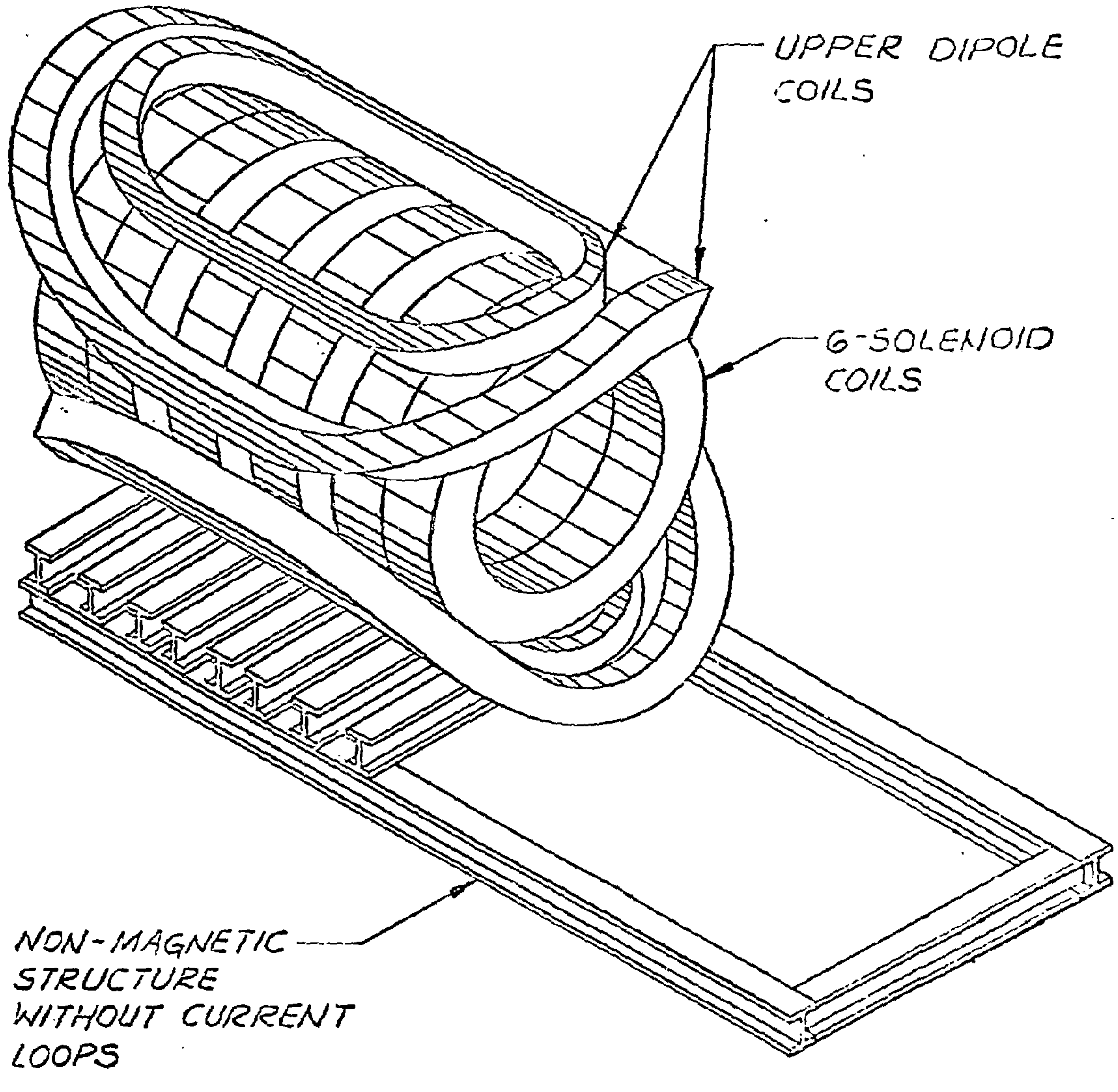


FIG. 3. SCHEMATIC DIAGRAM OF THE FW/B/S ELECTROMAGNETIC EFFESTS TEST STAND.

TABLE II

COIL PARAMETERS

	Solenoid Per Coil	Half a Dipole Inner	Half a Dipole Outer
Field Strength (T)	1.0	0.5	
Inner Radius (cm)	55.0	80.0	80.0
Outer Radius (cm)	75.0	100.0	100.0
Axial Length (cm)	30.0	280.0	320.0
Azimuthal Length (cm)	--	12.0	19.0
Angle to Center (deg.)	--	54.0	18.0
NI (kiloampere turns)	417.0	247.2	395.2
Stored Energy (KJ)	254.5	537.0	
Inductance L (mH)	27.0	28.0[a]	
Resistance R (mΩ)	17.2	39.5[a]	
Peak Current (kA)	4.4	6.2[a]	
DC Voltage Drop (V)	75.68	245.0[a]	
Power Consumption (KW at 1 ppm)	55.5	95.7	

[a]One inner and one outer coil in series

broad range of fusion reactor components. These two contractors collaborate very closely in their efforts. An outline of activities planned for the next two to three years is given in Fig. 4.

3.0 BREEDER BLANKET DEVELOPMENT

Successful and timely development of tritium breeder blanket technology requires an integrated approach consisting of three essential elements; namely: (1) design and modelling activities, (2) measurement of materials properties, and (3) engineering testing and simulation. At ANL, these three elements are being developed in parallel with effective communication and exchange between all of the elements.

The first element is represented by a comprehensive blanket design effort, which involves several manyears of work at ANL and elsewhere. This effort is highlighted by the STARFIRE design. [5] The design and modelling activities serve to: (1) critically assess the status of understanding of breeder blanket technology, (2) prioritize critical data needs and areas for study, and (3) provide a model which serves as a theoretical framework for understanding the implications of experimental results. Experimental programs involving ANL and other organizations, which are designed to develop the required materials properties data base, are discussed below in Sections 3.1 and 3.2. Blanket engineering development and testing, as represented by the TRIO program, is discussed in Section 3.3.

3.1 Materials studies

Current activities in breeder blanket development have given increasing attention to the lithium-containing ceramic materials to assess their suitability for use in breeder blanket designs. Materials under consideration include Li_2O, $LiAlO_2$, Li_4SiO_4, Li_2SiO_3, Li_2ZrO_3, and Li_2TiO_3. These materials are being investigated because of their high lithium-atom density; attractive chemical, physical, and mechanical properties; and thermal stability. The data base for the lithium-containing ceramics, the subject of a recent workshop, [6] is not large, but the inherent safety advantages of these materials provide strong incentives for consideration as candidate breeder materials. The goal of current solid breeder programs is to develop the requisite data base for candidate materials so that appropriate breeder concepts can be developed.

Characterization of the candidate solid breeder materials will give attention to each materials behavior as regards chemical stability, thermal properties, tritium release characteristics, preparation and fabricability, neutron response, and radiation effects. In each material, tritium is produced primarily by the reaction $^6Li(n,\alpha)^3T$. Tritium is likely to leave the breeder as T_2 or T_2O into a gas purge stream. Basic to a firm understanding of this mechanism are data on tritium solubility, phase relationships, and vaporization behavior of the tritium solid breeder system.

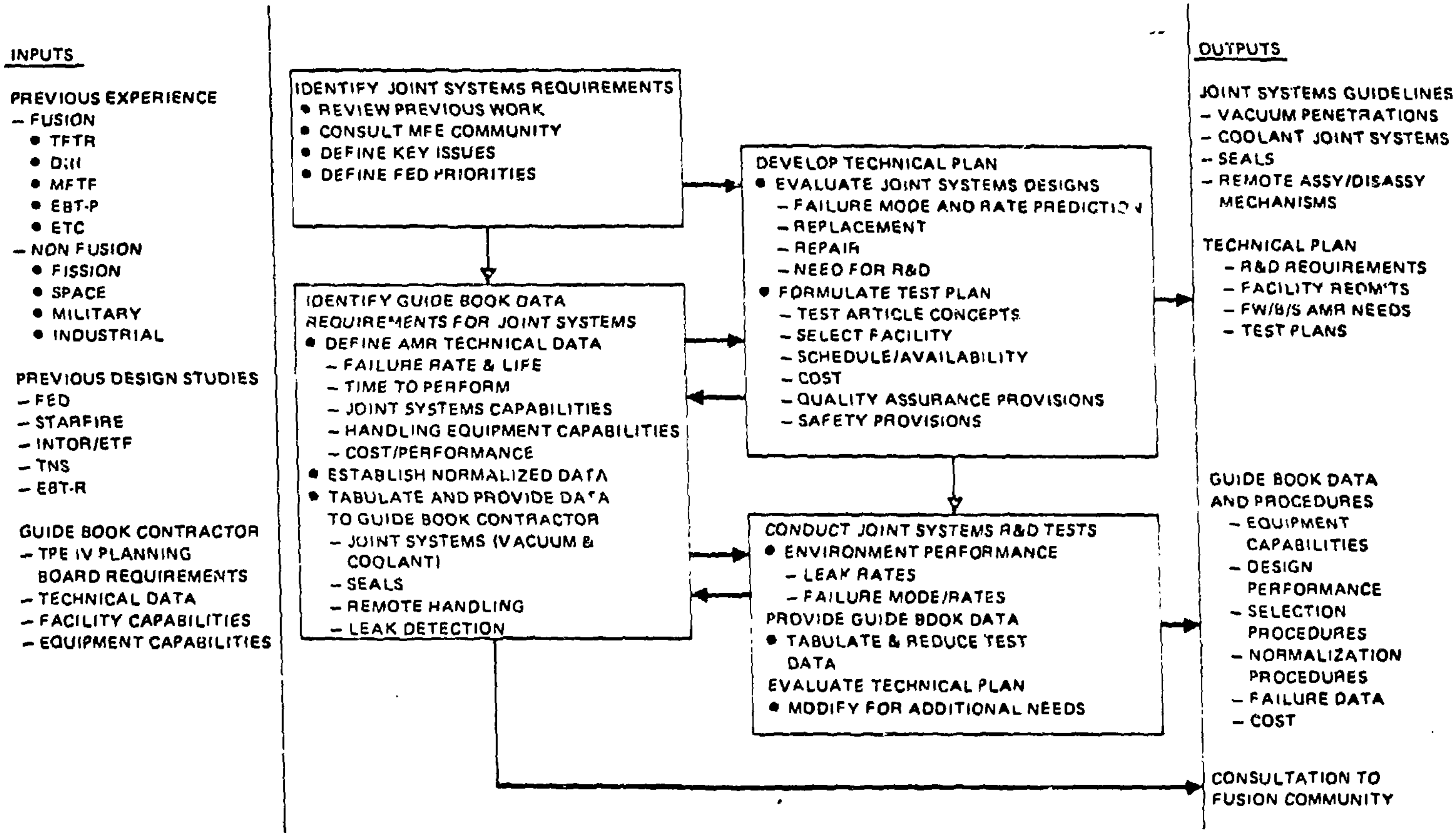


FIG. 4. AMR ACTIVITIES COVERED BY TPE-IV.

Thermochemical and thermophysical properties data are particularly critical for the evaluation of candidate breeder materials. Briefly discussed are recent measurements relevant to this important issue.

The thermal conductivity of lithium ceramics is important not only for the engineering application of these materials in a fusion environment but also for the effective designing of irradiation experiments for these materials. There is concern that too low a thermal conductivity in these materials would lead to excessive temperatures or require excess cooling to keep the system under control. In the irradiation testing of these materials, thermal conductivity is a key property that determines each material's performance.

The thermal conductivity of candidate solid breeder materials is given in Table III. A least squares analysis was taken on reported data to give an expression for theoretically dense material. Takahashi has reported [7] thermal conductivity data for Li_2O as a function of temperature and density. The data of Gurwell [8] was on material that was a mixture of the γ and α allotropes of $LiAlO_2$.

At low temperatures (~400°C) the thermal conductivity of Li_2O is almost twice that of $LiAlO_2$ although at higher temperatures (>700°C) the thermal conductivity of Li_2O approaches that of $LiAlO_2$. The data for Li_4SiO_4 was found to be similar to that of $LiAlO_2$ throughout the range of temperatures measured. The thermal conductivity for Li_2ZrO_3 appears to be lowest for all materials under consideration.

Specific heat measurements were made for candidate solid breeder materials and data are given in Table IV. The specific heat of each of the materials increases with temperature, with Li_2ZrO_3 possessing the lowest slope. Theoretically, the specific heat of a material should start near zero at absolute zero and increase to a plateau value at high temperature. The upper limit is approximately 3 R (three vibrational degrees of freedom). The predicted 3 R limits coincides reasonably well with the data at high temperatures.

The enhanced volatility of Li_2O in the presence of moisture has been noted by Brewer and Margrave [12], van Arkel, et al [13], and by Berkowitz-Mattuch and Buchler [14]. For the needs of designers, these papers are of limited use. Recent gas-solid equilibrium studies have provided a more quantitative understanding of the vaporization phenomena. With water vapor in the carrier gas, LiOH(g) was the primary gaseous species according to the reaction LiO(s) + H O(g) → 2LiOH(g). The results of our investigations can be expressed as log k (eq) = 10.101 - 1.8463.

In the absence of water vapor, the total pressure of lithium-bearing species was derived from the mass spectrometric measurements of Kudo, et al [15]. The gaseous species Li(g), Li_2O(g), Li_2(g), and Li_2O_2(g) were found with Li(g) and Li_2O(g) being predominant. Intercomparison of the data of Kudo, et al, with that presented here clearly shows that moisture has a very significant effect on enhancing the volatibility of lithium oxide.

TABLE III

THERMAL CONDUCTIVITY OF LITHIUM CERAMICS

$$K = K_o + \frac{1 - P^a}{1 + BP}$$

Material	K_o (w/cw K)	B	Ref.
Li_2O	$\frac{1}{1.40 + 0.01828T}$	$1.95 - 8 \times 10^{-4}T$	[7]
$LiAlO_2$	$0.0147 + 9.43/T$	$1.95 - 8 \times 10^{-4}T$	[8]
$LiAlO_2$	$0.00886 + 19.12/T$	$1.95 - 8 \times 10^{-4}T$	[9]
Li_4SiO_4	$0.0198 + 8.5/T$	$1.95 - 8 \times 10^{-4}T$	[9]
Li_2ZrO_3	$0.0102 + 6.68/T$	$1.95 - 8 \times 10^{-4}T$	[9]

^{a}P = porosity of solid

TABLE IV

SPECIFIC HEAT OF LITHIUM CERAMICS

Material	Heat Capacity	Ref.
Li_2O	$0.6014 + 7.95 \times 10^{-5}\ T - 2.00\ Z \times 10^{4}\ T^{-2}$	[10]
$LiAlO_2$	$0.335 + 4.39 \times 10^{-5}\ T - 1.0 \times 10^{2}\ T^{-2}$	[11]
$LiAlO_2$	$0.250 + 9.6 \times 10^{-5}\ T - 4.30 \times 10^{3}\ T^{-2}$	[9]
Li_4SiO_4	$0.224 + 3.4817 \times 10^{-7}\ T + 9.58 \times 10^{2}\ T^{-2}$	[9]
Li_2ZrO_3	$0.179 + 7.45 \times 10^{-5}\ T + 1.1263\ T^{-2}$	[9]

Complementary to this effort at ANL is the initiation of irradiation experiments to enlarge our experience on the behavior of candidate materials in a neutron environment. The DOE/OFE, under ANL supervision, is sponsoring two irradiation experiments. One experiment (TRIO) will be carried out in the Oak Ridge Reactor (ORR) and is to be a dynamic study of tritium recovery from γ-$LiAlO_2$. This is discussed in further detail in Section 3.3. The other experiment (FUBR) is being carried out by the Hanford Engineering Development Laboratory (HEDL) and is a scoping study of four candidate materials (Li_2O, γ-$LiAlO_2$, Li_4SiO_4, and Li_2ZrO_3). FUBR will be carried out in the Experimental Breeder Reactor-II (EBR-II). This latter study is significant in that it will be the first time these materials will be exposed to a neutron environment prototypical of that expected in a fusion reactor.

Studies have also been initiated at ANL on measurements of the corrosive behavior of candidate solid breeder materials with various structural alloys using the reaction couple method. Specimen tabs of the structural alloys 316-SS, Inconel 625, Ti6242, and HT-9 were exposed at 600°C to Li_2O, $LiAlO_2$, and Li_2SiO_3. Corrosion-product layers contained elements present both in the alloy and the ceramic. Several mixed-oxide reaction products ($Li_xM_yO_2$, M = Ni, Fe, Cr) were identified. Quantitative information on corrosion rates, corrosion-layer thickness, and corrosion-layer chemical composition is presently being developed.

3.2 Japanese collaboration

An experimental program was conducted in collaboration with the staff at the Japan Atomic Energy Research Institute (JAERI) during 1981 with the purpose of expanding the engineering data base on two ternary lithium oxides, Li_2ZrO_3 and Li_2TiO_3. The subjects studied were thermal diffusivity, compatibility, and tritium release behavior after irradiation. Material from Alfa, Inc., after heat treatment to remove LiOH and Li_2CO_3 impurities, was cold pressed and sintered to produce discs 66-90% theoretical density (TD) for Li_2ZrO_3 and 70-80% TD for Li_2TiO_3.

The thermal diffusivity, α, was measured by the laser pulse method from $\sim$450-1100 K for both materials for a number of densities. Using literature values for the heat capacity, the thermal conductivity was calculated. Both materials had low thermal conductivities, 1-2 W/mK for Li_2ZrO_3 and $\sim$2 W/mK for Li_2TiO_3. Because hysteresis was observed during the measurement of the thermal diffusivity which may have been caused by microcracking during thermal cycling, attention must be given to the effects of thermal cycling for these materials and, perhaps, other ternary oxides.

Compatibility tests were conducted for the oxide/alloy couples composed of 75% TD Li_2TiO_3, 88% TD Li_2ZrO_3 or 86% TD Li_2O and HT-9, 316-SS or 316-SS coated with 20 μm of nickel in helium environments for 410 h at 923 K and 1035 h at 823 K. Appreciable reaction, i.e., formation of scale, was noted for Li_2O/HT-9, Li_2O/316-SS, and

and Li_2ZrO_3/HT-9. For the conditions studied, the remaining couples appeared to form small amounts of scale. A noteworthy point is that for all reaction couples, even those with a nickel coating, the species $LiCrO_2$ was formed either at the scale surface (Li_2TiO_3) or under a layer of Li_5FeO_4 (most Li_2O samples) or under a layer of $LiFeC_2$ (most Li_2ZrO_3 samples). The $LiCrO_2$ may serve as an inhibitor for further reaction when it forms at the scale surface.

The tritium release behavior of sintered Li_2TiO_3 and Li_2ZrO_3 irradiated for $\sim 10^{15}$, 10^{16}, and 10^{17} n/cm^2 was examined using constant rate heating and a flowing gas system. For both Li_2TiO_3 and Li_2ZrO_3 the major species was HTO, but there may have been present significant amounts of HT. (Experimental problems precluded the resolution of this issue.) For Li_2ZrO_3, two release peaks were observed, located at 695 K and 770 K for 10^{17} n/cm^2 samples heated at 1 K/min. under hydrogen. These peaks were located at higher temperature (1) if faster heating rates were used, (2) if helium gas was used, or (3) if the samples were exposed to water. Samples receiving lower fluences had two maxima at lower temperatures. For Li_2TiO_3, two release peaks were also observed, but the one at the higher temperature was significantly smaller. The maxima were at 634 K and 743 K at 1 K/min. using hydrogen. The peaks were shifted to higher temperatures (1) if faster heating rates were used or (2) if helium gas was used. Exposure to water resulted in a tailing effect on the peaks. Changes with fluence were not significant.

After being heated to 1173 K, the irradiated samples were dissolved to determine the tritium residue. The Li_2ZrO_3 samples contained a γ-contaminant; therefore, only an upper limit is available (<0.02 wpm). For Li_2TiO_3, the residue was 0.0003 wpm.

To summarize, both Li_2ZrO_3 and Li_2TiO_3 have low thermal conductivity ($\leq$2 W/mK). In addition, thermal cycling may introduce microcracks which would serve to enhance or decrease their thermal conductivities. The Li_2TiO_3 was compatible with all alloys tested. The Li_2ZrO_3 seems to have significant penetration with HT-9. Both materials release HTO and HT. The Li_2ZrO_3 releases a significant tritium fraction at temperatures $\geq$770 K. The Li_2TiO_3 releases most tritium at temperatures $\leq$743 K. In addition, Li_2TiO_3 release behavior did not appear to be sensitive to neutron fluences up to 10^{17} n/cm^2.

3.3 TRIO-01: An in-pile test of *in-situ* recovery of tritium from a solid breeder

TRIO-01 is a comprehensive in-pile test of *in-situ* tritium recovery and thermal-hydraulic performance of a miniaturized blanket assembly. This experiment is to begin in 1982 in the ORR at ORNL. As previously discussed, design and modelling activities such as STARFIRE [5], have provided a starting point for the design of this experiment.

The configuration [5] of the solid breeder blanket is illustrated in Fig. 5. Tritium bred within grains must find its way out of the solid into the purge stream, where it is swept away for recovery. A number of specific steps required for recovery have been identified, including: (1) bulk diffusion of tritium from the interior of grains to grain edges; (2) desorption of tritium (T_2O) at grain edges; (3) migration of tritium along grain boundaries; (4) "permeation" through small pores in the pellets (see Fig. 5); (5) "percolation" through channels between the pellets; and (6) convective mass transfer of tritium through the purge stream out of the blanket. The tritium inventory in the blanket depends upon the rates of these mechanisms. The rates of all these mechanisms have not been quantitatively determined at this time, although it can be said that the six steps will depend upon a number of kinetic and thermodynamic factors. The primary purpose of the TRIO-01 experiment is to determine rates of tritium release as a function of temperature and as a function of purge gas flow rate and composition. This information is expected to provide an understanding of the relative importance of various proposed kinetic mechanisms, e.g., the six mechanistic steps outlined above.

The TRIO-01 capsule design is shown in Fig. 6. The breeder is in the form of a hollow cylinder, and the bred tritium is removed by a sweep gas. The configuration is meant to provide a controlled simulation of conditions in a fusion reactor blanket. As shown in Fig. 6, temperature control is achieved by a combination of a gas gap and heaters. The key experimental parameters are listed in Table V.

The calculated tritium production and heating rates are shown in Table VI. Because the burnup of ^6Li is about 50% for every 30 days of irradiation in ORR, nuclear heating rates and tritium production rates will change during the course of the experiment, as illustrated in Fig. 7.

Because of the importance of temperature and temperature gradients to understanding tritium release, control and measurement of temperature is an essential part of this experiment. The temperature gradients for hollow cylinders of $LiAlO_2$ under the anticipated experimental conditions for TRIO-01 are shown in Fig. 8. It can be seen that the reference configuration, having an outer radius of 1/2 in. and an inner radius of 1/4 in. as shown in Fig. 6, has a temperature gradient (ΔT) of about 150°C. Because temperature and temperature gradients will be monitored and controlled, the experiment will also be a test of thermal-hydraulic performance of solid breeders.

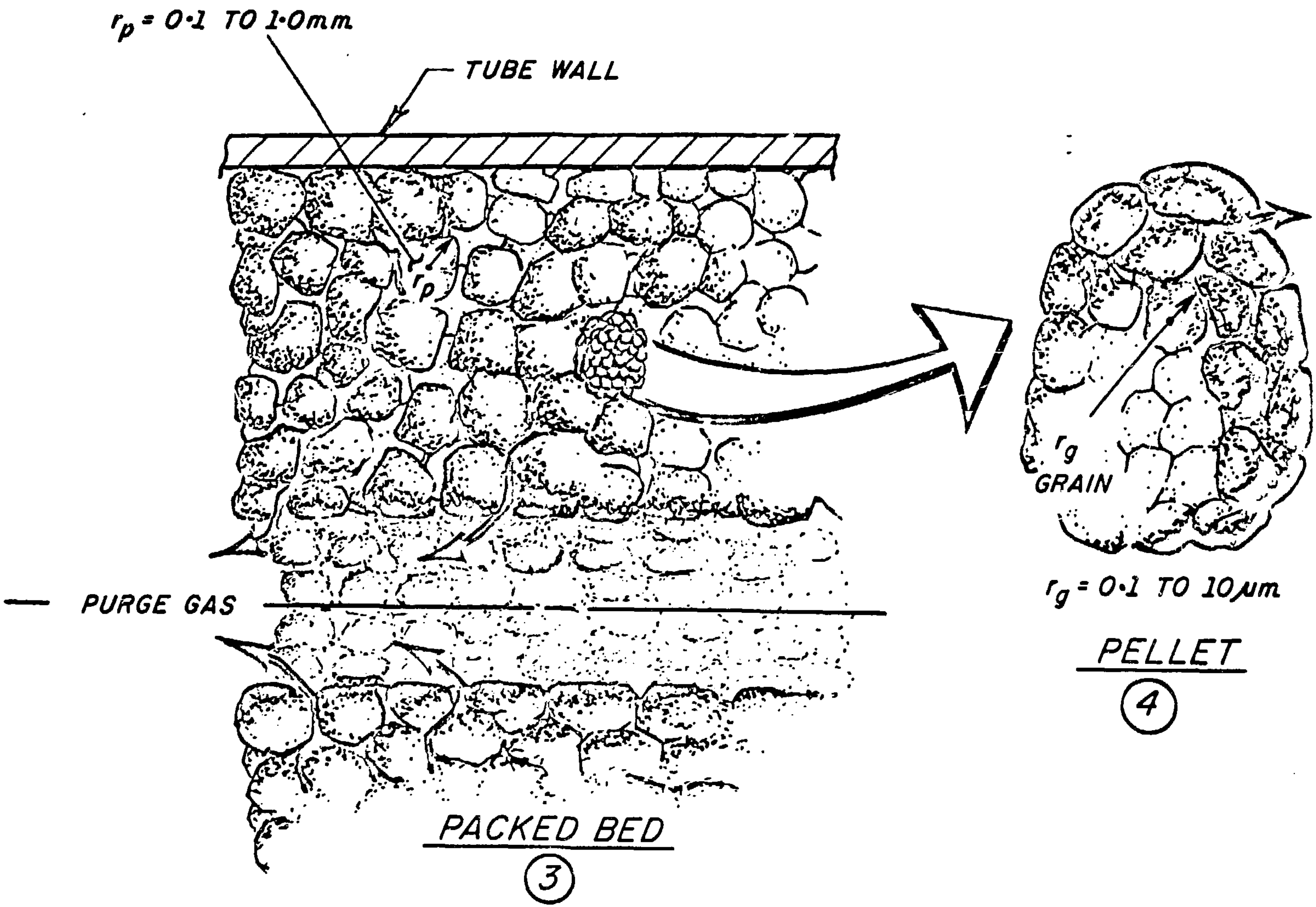


FIG. 5. TRIO-01 SOLID BREEDER BLANKET CONFIGURATION.

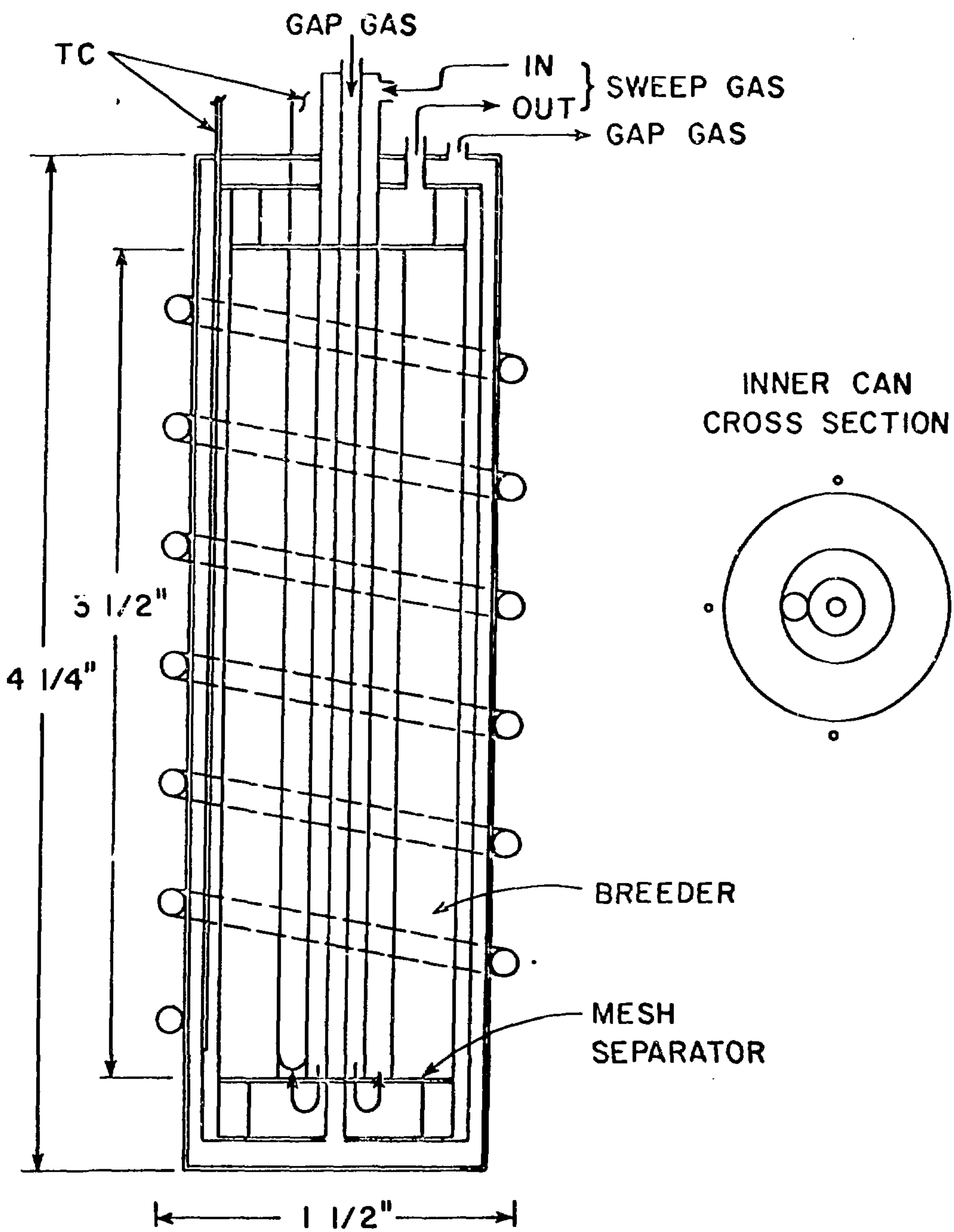


FIG. 6. TRIO-01 CAPSULE CONFIGURATION.

TABLE V

REFERENCE DESIGN PARAMETERS FOR THE TRIO-01 EXPERIMENT

BREEDER

γ-LiAlO

Theoretical Density: 60%

Grain Size: 0.3 μm

Thermal Conductivity: ∿1.4 w/m•K

TEMPERATURE

Range: 400°C to 1000°C

Gradient: ∿100°C or less

ISOTROPY

0.5% ^{6}Li

99.5% ^{7}Li

PURGE GAS

He, plus H_2, D_2, or O_2

100-1000 cm^3/min. (variable)

MONITORING

Temperature and Temperature Gradients

Tritium and Chemical Species in Purge Gas

Neutron Flux by Real-Time and Dosimetry Methods

TABLE VI

TRITIUM PRODUCTION AND HEATING RATES
IN TRIO-01 EXPERIMENT

INITIAL TRITIUM PRODUCTION RATE $\dot{T}$

r (mm)	$\dot{T}$ atoms/s·cm^3 x 10^{13}
12.7	2.13
10.7	2.11
8.75	2.10
6.77	2.09

HEATING RATES (W/cm^3)

n	γ	Total
6.3	4.2	10.5

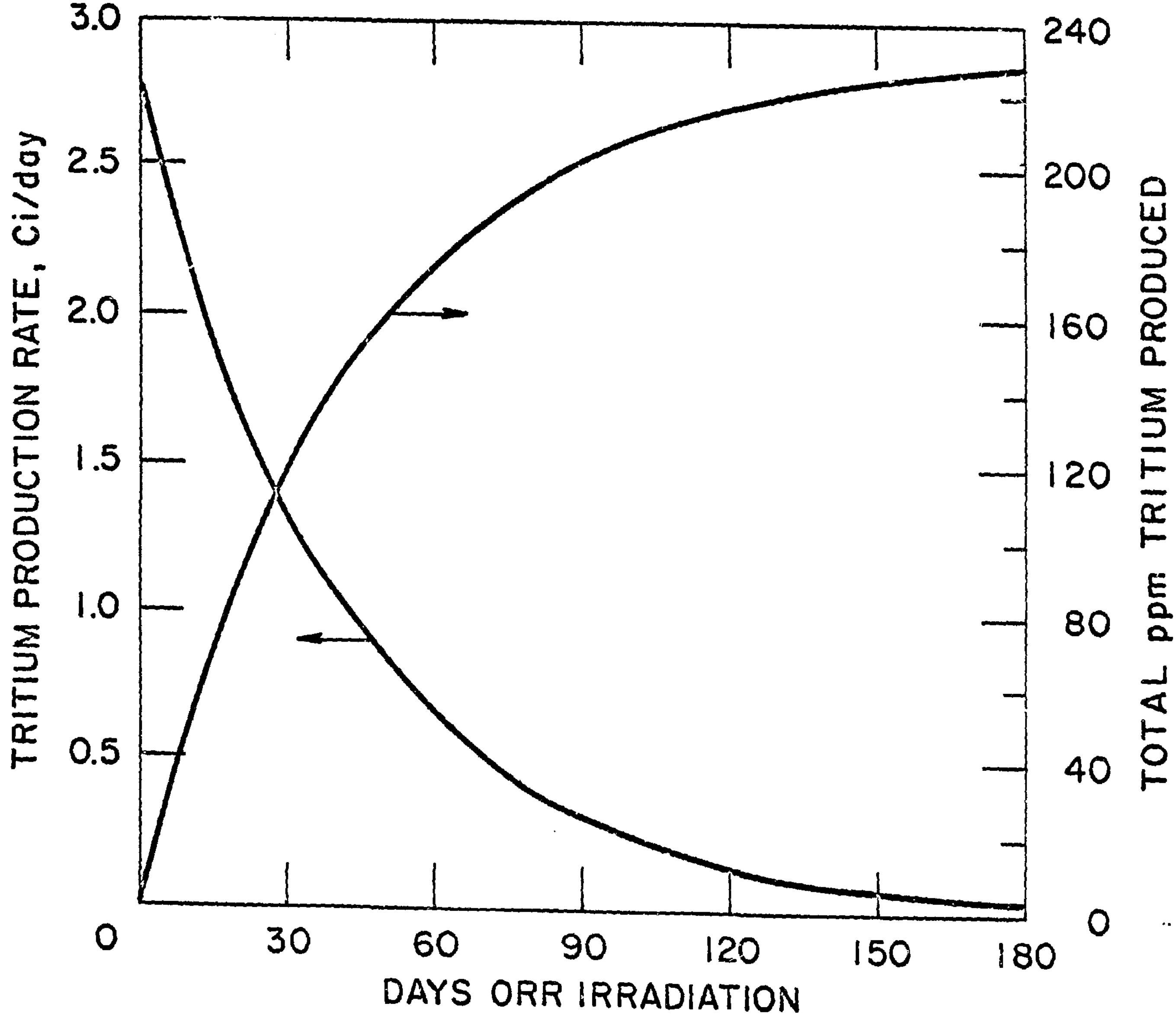
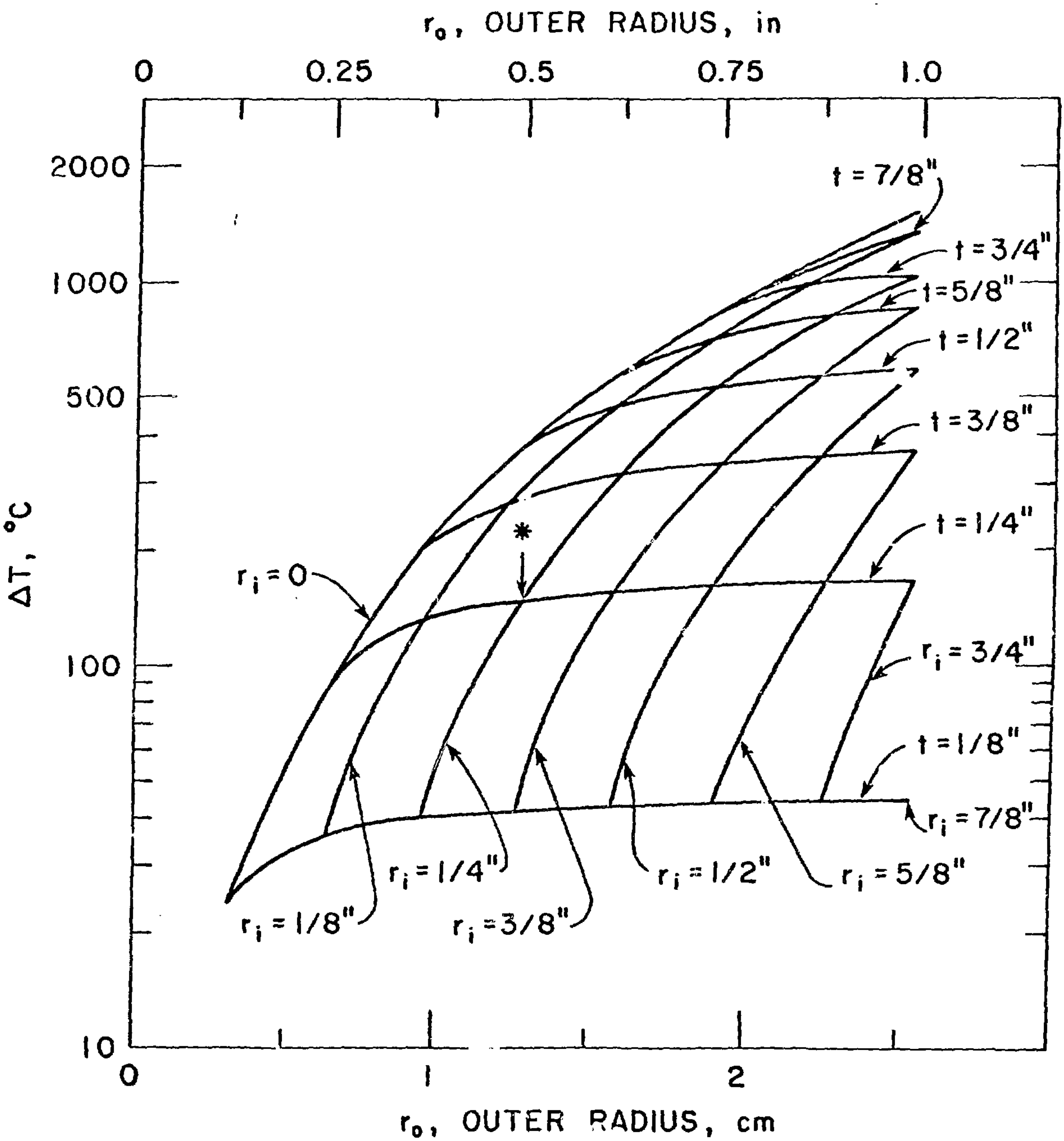


FIG. 7. TRITIUM PRODUCTION IN TRIO-01.



* = REFERENCE   t = THICKNESS r_i = INSIDE RADIUS

$\dddot{q}$ = 10.5 w/cm^3 κ = 1.16 J/m · k

FIG. 8. THERMAL-HYDRAULIC ANALYSIS: ΔT IN TRIO-01 BREEDER.

ACKNOWLEDGEMENTS

We would like to acknowledge the valuable input of the following people to the programs cited in this paper: (1) First Wall/ Blanket/Shield Engineering Test Program - T. Varljen, R. Rose, and J. Chi from Westinghouse Electric Corporation (TPE-I); K. Schultz from General Atomic Company and P. Hsu from EG&G Idaho, Inc. (TPE-II); W. Praeg from ANL (TPE-III); and J. White from Remote Technology Corporation and H. S. Zahn from McDonnell Douglas Astronautics Company (TPE-IV). (2) Materials Studies - G. W. Hollenberg from Hanford Engineering Development Laboratory, J. W. Davis from McDonnell Douglas Astronautics Company, and M. Tetenbaum from ANL. (3) Japanese Collaboration - All staff at JAERI, especially T. Kurasawa, K. Noda, T. Takahashi, H. Takeshita, T. Tanifuji, and H. Watanabe. (4) TRIO-01 Experiment - J. Scott and K. Thoms from Oak Ridge National Laboratory, R. Wiswall from Brookhaven National Laboratory, K. Schultz from General Atomic Company, and D. Smith and R. Arons from ANL.

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FIGURE CAPTIONS