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THE FIRST WALL, BLANKET, SHIELD ENGINEERING TECHNOLOGY PROGRAM

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ABSTRACT

The First Wall/Blanket/Shield Engineering Technology Program sponsored by the Office of Fusion Energy of DOE has the overall objective of providing engineering data that will define performance parameters for nuclear systems in advanced fusion reactors. The program comprises testing and the development of computational tools in four areas: (1) Thermomechanical and thermal-hydraulic performance of first-wall component facsimiles with emphasis on surface heat loads; (2) Thermomechanical and thermal-hydraulic performance of blanket and shield component facsimilies with emphasis on bulk heating; (3) Electromagnetic effects in first wall, blanket, and shield component facsimilies with emphasis on transient field penetration and eddy-current effects; (4) Assembly, maintenance and repair with emphasis on remote-handling techniques. This paper will focus on elements 2 and 4 above and, in keeping with the conference participation from both fusion and fission programs, will emphasize potential interfaces between fusion technology and experience in the fission industry.

FIRST WALL, BLANKET AND SHIELD - THE FUSION REACTOR NUCLEAR SYSTEM

Together the first wall, blanket and shield provide a nuclear envelope that isolates the fusion plasma and the energetic particles produced by the plasma from the rest of the fusion reactor. Figures 1 and 2 show these components in STARFIRE, a conceptual design for a commercial tokamak fusion reactor and in the mirror reactor described in MARS, the Mirror Advanced Reactor Study. Figure 3 shows details of the first wall, blanket and shield in the STARFIRE design.

First-wall components will directly face the plasma and be subjected to a severe environment including intense electromagnetic radiation and bombardment by high energy neutrons and energetic particles from the plasma. The primary function of the first wall is to maintain the physical boundary that defines the plasma chamber without adversely affecting the plasma (by introducing impurities). Rejection of the intense surface heat loads is an important requirement in fulfilling this function in tokamaks (mirror devices have much lower first wall heating) and particularly in specialized first wall components (limiters, armor and divertor plates) designed to protect the

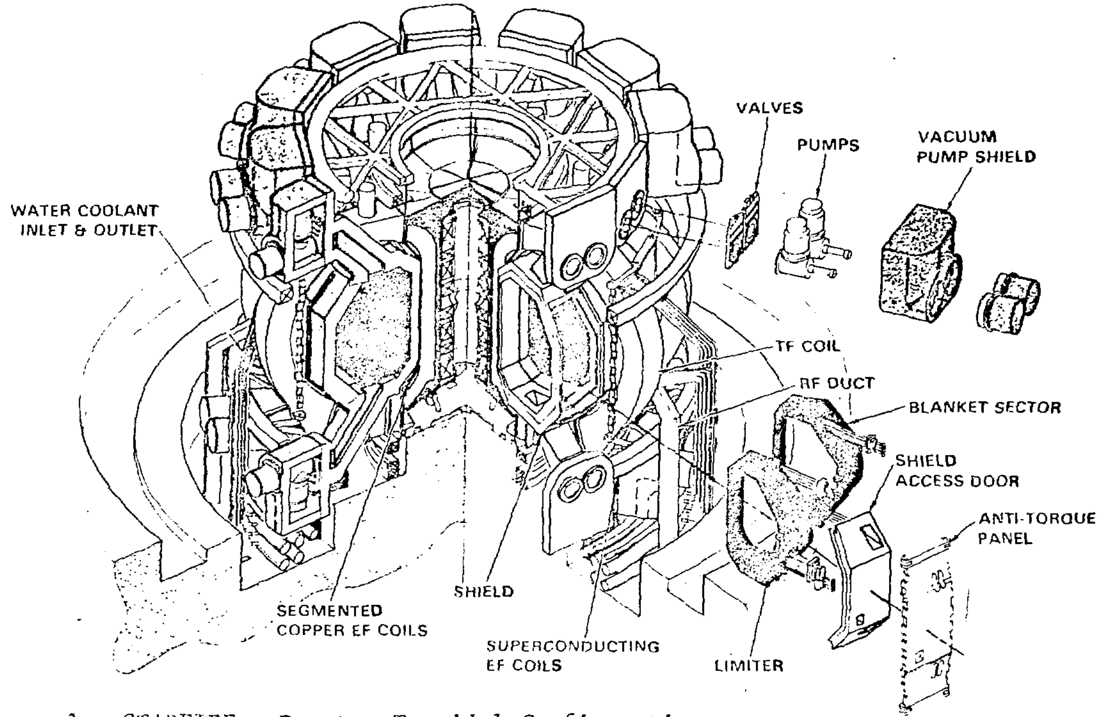

Figure 1. STARFIRE - Reactor Toroidal Configuration.

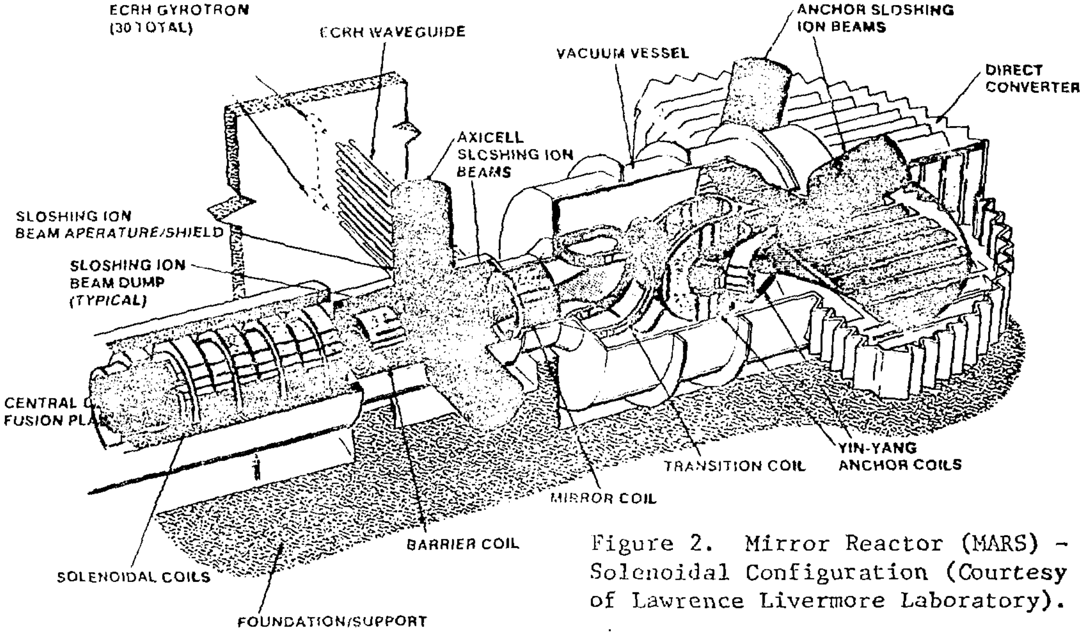

Figure 2. Mirror Reactor (MARS) - Solenoidal Configuration (Courtesy of Lawrence Livermore Laboratory).

first wall by preferentially interecepting plasma particles. In a fusion reactor that produces 1000 MW of power from the fusion reaction ($D + T \rightarrow He + n$, 17.6 MeV), the firstwall components will receive about 200 MW as surface heat.

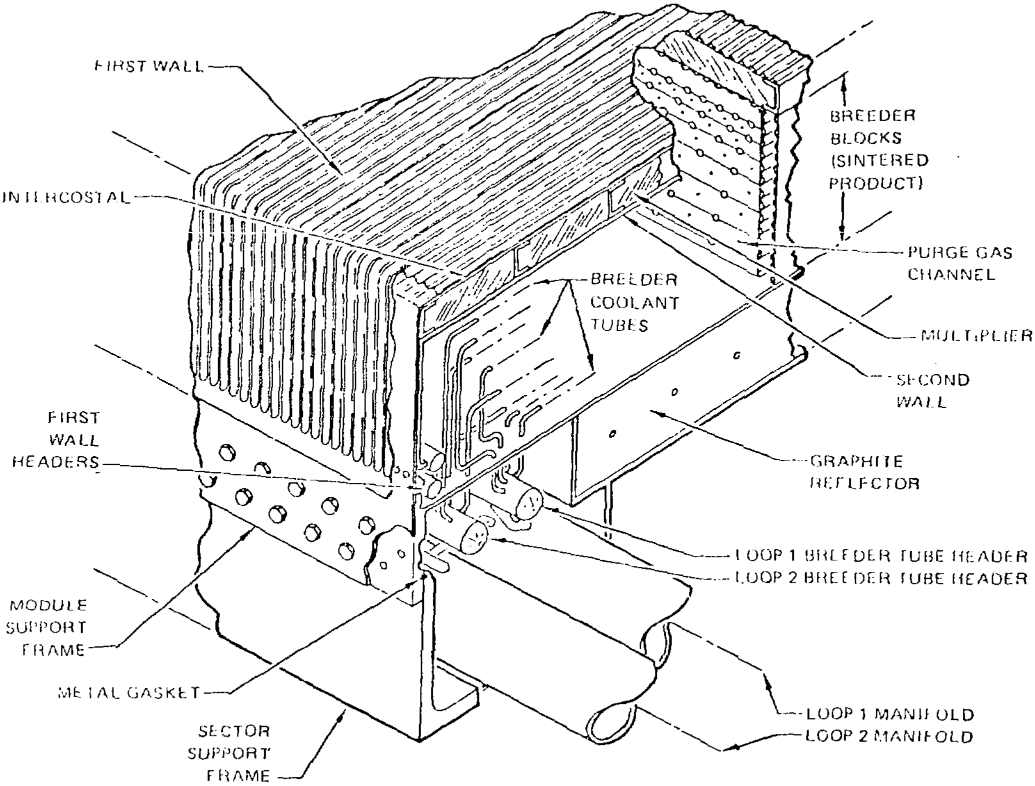

Figure 3. Details of the STARFIRE Features of the First Wall and (Solid Breeder) Blanket.

The blanket has two functions, generating heat and producing tritium to fuel the reactor, and must provide the extraction of sensible heat and tritium from the blanket. Heat generation in the blanket is dominated by the nuclear reactions through which lithium (^6Li and ^7Li) transmutes into tritium and helium. Liquid lithium compounds such as lithium itself or Pb-Li, and solids for example oxides such as Li_2O and $LiAlO_2$, are both actively being investigated as breeding materials. Many styles of blanket designs are considered potentially feasible.

The shield provides biological protection for personnel and reduces radiation and nuclear heating to levels acceptable for the operation of sensitive components such as superconducting magnets. Some of the primary shielding in most reactor designs is readily moveable to permit remote maintenance, assembly and repair.

THE FW/B/S PROGRAM

The ultimate goal of the First Wall/Blanket/Shield Engineering Technology Program (FW/B/S Program) is to provide, through engineering analyses and component tests, the support required to design and construct functional, reliable, maintainable, environmentally acceptable FW/B/S systems for magnetic fusion reactors. The data supplied by the FW/B/S Program will be integrated with technical input from related work in other parts of the fusion program to produce the necessary data base. The principal objectives of the experimental and analytical program area: (1) verify computational tools, (2) provide an engineering data base that includes preliminary engineering information on FW/B/S system performance and reliability and (3) provide engineering input to verify the integration of the total FW/B/S system and subsystems.

The organization of the FW/B/S Program is shown in Figure 4. Argonne National Laboratory provides technical direction of the program and performs

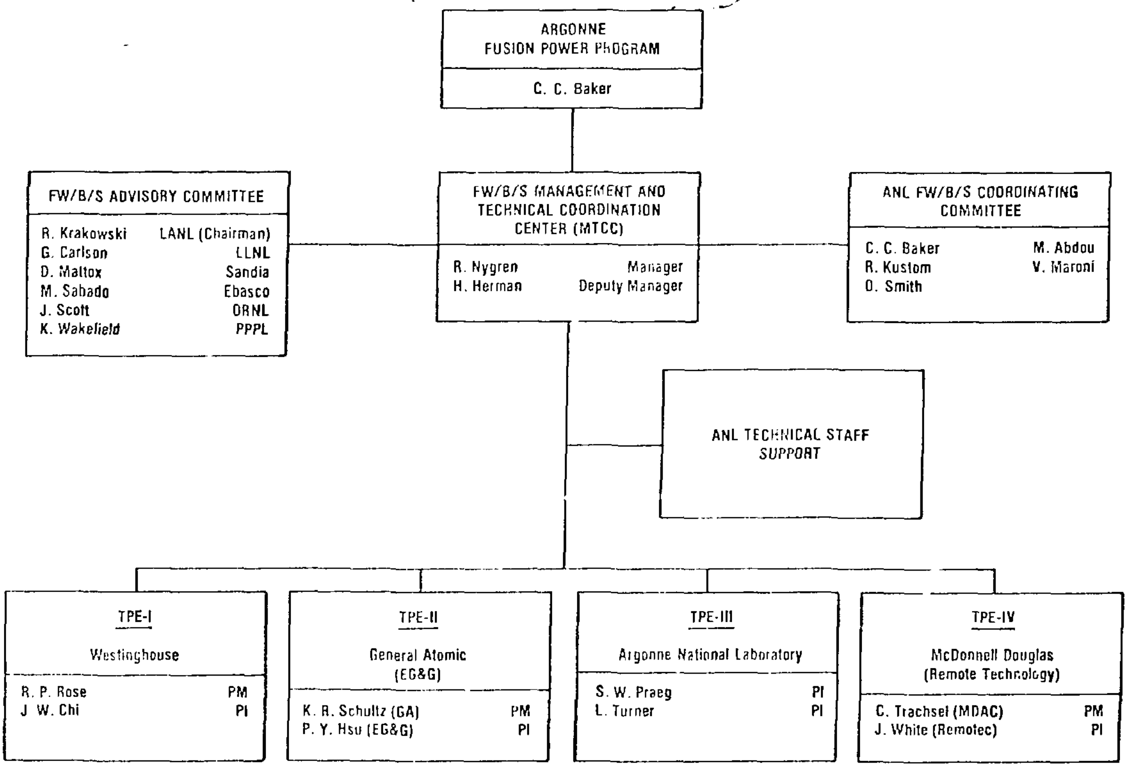

Figure 4. Organizational Structure of the FW/B/S Program

one of the four program elements. Industrial contractors perform the other three elements. Participation by industry in R&D for fusion technology is also one of the organizational objectives of this program.

Recently the program proceeded from Phase 0 (experimental planning) to Phase I (experiments and evaluation). Phase 0 culminated with the preparation of Detailed Technical Plans for each of the four test program elements (TPEs) and a revised Program Plan. Near term work involves the parallel development of experimental data and computational tools on specific engineering issues within four key areas of fusion technology related to FW/B/S systems, as described in the following sections.

TEST PROGRAM ELEMENT I

THERMOMECHANICAL AND THERMAL-HYDRAULIC TESTING OF FIRST WALL COMPONENT FACSIMILES WITH EMPHASIS ON SURFACE HEAT LOADS (WESTINGHOUSE ELECTRIC CORP.)

The first wall and other structures, such as limiters, which directly face the plasma must withstand large heat fluxes. One of the primary systems design problems for first wall components is adequate heat removal. Indicated in Fig. 5 are various surface heat loads expected on first wall components in (tokamak) fusion reactors. The axes in this figure are power and heated area, the two major parameters for characterizing heating sources.

Westinghouse began surface heating tests in December 1981 using a 50 kW e-beam source. Completion in mid-1982 of a more powerful 100 kW facility (ASURF) capable of accommodating test pieces of about one square meter in size is imminent and a future upgrade of ASURF to 1 MW is anticipated. The increases in power planned for these sources provides the capability to test larger test articles that more accurately simulate the large multi-channeled heat rejection panels used in first-wall, limiter and divertor/collector designs. The experimental data will support the development and verification of analytical models that can accurately describe the temperatures, gradients, distortions and hydraulic behavior of advanced heat rejection components. This work is the subject of another paper [1] in this conference.

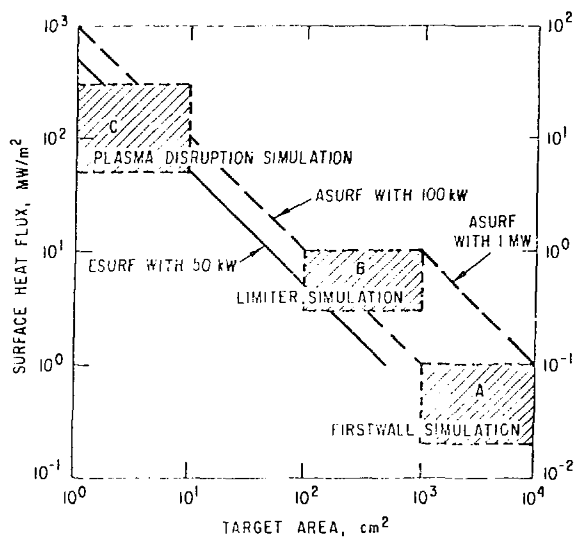

Figure 5. Parameters of surface heat flux and target area for ESURF and ASURF. Boxes show parameters appropriate for fusion simulations.

TEST PROGRAM ELEMENT II

THERMOMECHANICAL AND THERMAL-HYDRAULIC TESTING OF BLANKET AND SHIELD COMPONENT FACSIMILIES WITH EMPHASIS ON BULK HEATING (GENERAL ATOMIC CO. AND EG&C IDAHO).

The blanket in a fusion reactor will collect process heat and will produce tritium from lithium bearing materials. As a first step in establishing the detailed scope of TPE-II, various concepts for blankets and shields proposed to date were reviewed and broadly categorized according to basic styles of designs. Each general style of blanket and shield design was then evaluated with regard to various engineering concerns. The results of this evaluation are briefly summarized in Table 1.

Table 1 also mentions areas of concern that lie outside the purview of TPE-II, notably "materials concerns" and "tritium concerns". The ability to perform satisfactory analytical support for the testing in TPE-II depends heavily on adequate data in these areas of concern. Strong interfaces between the FW/B/S Program and other fusion programs, especially the US Fusion Reactor Materials Program, are anticipated.

This investigation [2] has led to two general conclusions. First, there was no clear winner among the blanket types; all showed promise in selected areas of performance and critical issues of concern were identified for each type. Second, the generic issues identified provide a useful basis for a test program. Typically the generic issues apply to one or even several but not all types of blankets. The near term objective in TPE-II is resolve blanket engineering issues on several fronts so that future comparisons of blanket types can be based on definitive evaluations of their projected performance.

The current focus in the FW/B/S Program, like the overall fusion program, is primarily on solid rather than liquid breeding materials. Most solid breeders currently being investigated are ceramic (lithium) compounds such as Li_2O and $LiAlO_2$. Limited data on these materials when applied to blanket designs for fusion reactors suggest that the useful temperature window for a given solid breeder will be fairly narrow. This constraint on operating temperature has been problemmatic in recent reactor design studies primarily because of lack of data on heat transfer characteristics and materials properties that are needed for predictions of the performance of breeder systems.

Figure 6 shows a simplified sketch of a small portion of a solid breeder blanket. The figure enumerates five features common to many designs. The first three features, (1) interface conductance, (2) heat sink i.e., coolant pipes and (3) bulk heat generation and bulk thermal conductances control the temperature at which the breeder operates. The purge flow channels (4) are passages through which helium flows and collects tritium that has migrated out of the solid breeder.

Table 1. TPE II Data Needs

DATA NEEDS	DESIGN OPTION: SHIELD I	SHIELD II	BLANKET I	BLANKET II	BLANKET III	BLANKET IV	BLANKET V
THERMAL-HYDRAULIC CONCERNS							
Thermal Conductivity		2	1	1			1
Thermal Contact Resistance		2	1	1			1
High Pressure Gap Flow				1			
Purge Flow Distribution		1	1				
Flow Distribution	3	3	2	1	2	1	1
Re-entrant Pressure Tube		3	1		1		
Heat Transfer Property Exchanges		2	1	1			2
Plenum Design	3	3	2	2	3	1	1
THERMOMECHANICAL CONCERNS							
Thermal Expansion Coefficient		2	1	2			3
Thermal Rartheting			1	2			
Thermal Stress		3	1	2			3
Module Joint Design	3	3	3	2	2	2	2
MATERIALS CONCERNS							
Materials Swelling	3	3	1	1			3
Coupled Swelling Creep and Embrittlement	3	3	2	1	2	3	3
Sintering			1	1			2
Compatibility			1	1	1	1	[illegible]
Vaporization			1	1	3	3	2
MHD CONCERNS							
Thermal-Hydraulic Effects					[illegible]	1	
Hydraulic Oscillations					3	2	
TRITIUM CONCERNS							
Tritium Release			1	1	3	3	2
Sintering Effects			1	1			3
Permeation			1	1	1	1	1

DESIGN OPTIONS

SHIELD I: Stainless steel with integral water cooling

II: Stainless steel/boron carbide composite with internal cooling tubes

BLANKET I: Low pressure solid breeder canister with coolant tubes

II: Clad solid breeder in high pressure module

III: Liquid metal breeder with coolant tubes

IV: Flowing liquid metal breeder

V: Mobile solid breeder

PRIORITY

1 = Critical Concern

2 = Concern

3 = Relevant but not expected to be of concern

Two critical engineering issues that pose potential problems for the exploitation of solid breeders are (1) the rate of heat transfer from a (ceramic) solid breeder to a (stainless steel) heat sink i.e., the "gap conductance problem" and (2) the long term configurational stability of a solid breeder operating at high temperature in a system with temperature gradients and flowing purge gas.

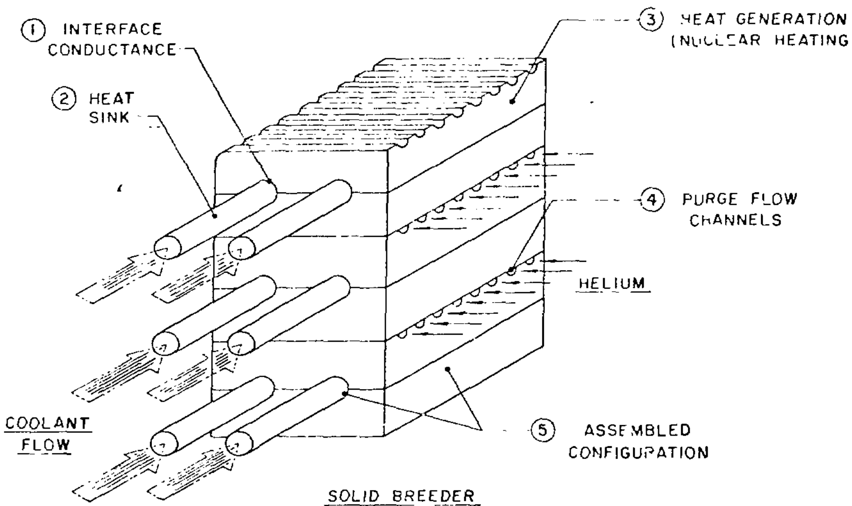

Figure 6. Sketch showing a small portion of a blanket with five features common to many blanket designs utilizing solid breeders.

The experimental program recently begun in TPE-II will produce preliminary results in 1982 from two types of scoping tests. In one type of test helium will pass through 2 mm holes in a series of Li_2O pellets held at high temperature (600°, 800° and 1000°C in separate isothermal tests). The objective is to demonstrate the configurational stability of the solid breeder during long term operational tests (about 1000 hours.). Some tests will also involve thermal cycling. The second type of test measures heat transfer across the interface between the solid breeder (Li_2O) and a stainless steel structure. Figure 7 is a simple sketch of the experimental concept and also shows the anticipated form of the temperature profile. Tests will be run for several gap thicknesses and for loading conditions in which the breeder and structure are pressed together. Both the absolute value of the conductance and the sensitivity to conditions at the interface are of interest.

Bulk heating of blanket and shield component facsimilies using nuclear heating in fission reactors was recognized at the outset as a potentially useful simulation. The basic phenomenon of nuclear heating is present in both the fission and fusion environments. The nuclear radiation also produces tritium in breeding materials and therefore such tests offer a more complete simulation of integrated performance. However the high cost of testing in fission reactors is an obvious practical constraint. Furthermore, extensively instrumented tests for studying particular aspects of performance such as heat transfer or thermohydraulics are more difficult to perform in fission reactor tests.

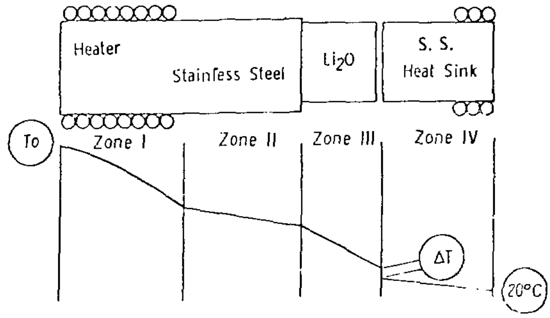

Figure 7. Simplified sketch showing basic elements in heat transfer test and the anticipated form of the temperature profile. (Courtesy of General Atomic Co.)

Plans for integrated testing of multiple effects, e.g., sintering, purge gas flow and temperature distributions are being developed as part of the ongoing development of testing strategy in TPE-II and the benefits of both nuclear and non-nuclear tests are being evaluated. These tests will complement in-reactor tests to study tritium production and radiation damage are already being performed in other programs (e.g., TRIO and FUBR).

TEST PROGRAM ELEMENT III

ELECTROMAGNETIC TESTING OF FIRST WALL, BLANKET AND SHIELD COMPONENT FACSIMILIES WITH EMPHASIS ON TRANSIENT FIELD PENETRATION AND EDDY CURRENT EFFECTS (ARGONNE NATIONAL LABORATORY).

In typical fusion reactor designs, the first wall, blanket and shield lie between the plasma and the magnets. Rapid changes in either the magnetic field or the plasma current result in eddy currents in the structure surrounding the plasma. The eddy currents can produce unwanted results of two types. First, if changes in the magnetic field are desired for either plasma control or ohmic heating, then the eddy currents degrade the field. Second, the eddy currents in some components can produce large forces and torques.

ANL is now constructing a magnet facility, the Fusion Electromagnetic Induction Experiment, FELIX, to study electromagnetic effects. Operation is expected in 1983. At present the calculational tools for analysis of electromagnetic effects in the segmented, inhomogeneous configurations of fusion reactors are simply not adequate. Using FELIX and test pieces with progressively more sophisticated configurations, calculational models appropriate for use by designers of fusion reactors will be developed. Descriptions of FELIX [3] and the test program [4] have been reported elsewhere.

TEST PROGRAM ELEMENT IV

STUDIES OF ASSEMBLY, MAINTENANCE AND REPAIR WITH EMPHASIS ON REMOTE HANDLING TECHNIQUES (MCDONNELL DOUGLAS ASTRONAUTICS CO. and REMOTE TECHNOLOGY).

Components in fusion reactors (first wall, blanket, shield, etc.) will become activated either directly from neutron radiation or indirectly from tritium or activated corrosion products transported by the coolant and thus remote maintenance and repair will be necessary. The range of operations to be performed remotely is large, from the lifting and transport of reactor segments weighing perhaps hundreds of tons to the delicate and precise work of coupling electrical leads or locating vacuum leaks. The potential scope of this subject is vast and many undertakings would imply large scale efforts beyond the current scope of this program. However important advances are possible even with limited resources.

The general strategy for TPE-IV has been to identify generic issues of concern where a modest development in hardware could significantly advance the "demonstrated technology" available to designers. Some examples of generic problem areas where significant benefits from modest development of hardware can be anticipated are: (1) leak detection and location; (2) electrical fault detection, location and damage assessment; (3) remote viewing and monitoring for physical surveillance and damage assessment; (4) remote mechanical connectors for vacuum seals; and (5) remote high-current electrical contacts for first wall sectors. Work in this program is already proceding on the latter two items above [5]. In parallel with the hardware development, there is also an effort to compile useful information for designers into a Designers' Guidebook on remote assembly, maintenance and repair.

Developing a clamping mechanism for remotely coupled (metallic) seals is one example of ongoing work in TPE IV. There are a variety of large (> 1 m^2) non-circular vacuum closures anticipated in fusion devices. The largest such closures are the joints between sectors of the devices, as shown in Figure 8. There are also large ports associated with vacuum pumping ports, neutral beam ports and in test devices like FED and INTOR penetrations roughly one meter square are included for experiment modules. Welding has been the commonly accepted design solution for large remote vacuum closures. Remote welding of large non-circular joints is not without problems but no credible alternative has yet emerged. In the FW/B/S Program, remotely actuated mechanical fasteners for use with metallic seals will be developed as a possible alternative. Early work will focus on the conceptual design of a clamping mechanism that offers promise for high reliability. Subsequent experimental studies will test the operation, sealing and reliability of the clamping mechanism.

OTHER INTERFACES WITH FISSION INDUSTRY

The following brief discussions gives three examples of subjects where experience from the fission industry will provide useful direct applications

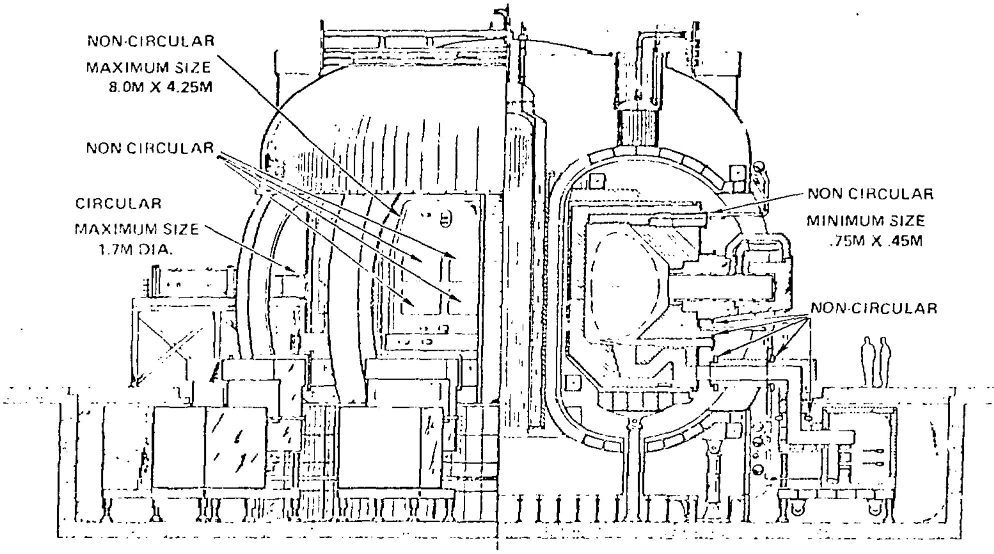

Figure 8. Fusion Engineering Device (FED)--drawing indicates several locations where large seals are required.

to fusion or, at minimum, useful starting points for development. The three examples are: remote handling, transport of radioactive (corrosion) products and radiation-hardened instrumentation. For more information on these subjects the reader is refered to major conferences on fusion engineering [6, 7] and project design reports [8-10].

REMOTE HANDLING

For magnetically confined fusion devices, the different types of hardware, notably the preponderance of electrical equipment and the somewhat complicated configurations, especially for tokamaks, make the requirements for remote handling for fusion plants more vast than for fission plants. One obvious difference is in the size and weight of components. For example, the estimate weight of one sector of the blanket in STARFIRE is 65 tons and the shield door is 179 tons. For large component handling the most directly applicable technology from the fission industry probably lies in fuel reprocessing plants and in the NIRVA (nuclear rocket) program.

TRANSPORT OF ACTIVATED (CORROSION) PRODUCTS

The migration of radioactive material from components in the nuclear island (surrounding the plasma) through the coolant to plant equipment, such as heat exchangers and valves, where the accumulation of activated material can complicate maintenance by limiting the access of personnel and making procedures for repair and replacement much more time consuming. The coolants being considered in fusion devices, i.e., water, liquid metal (e.g., as Li or Li-Pb) and helium, have also been investigated in the fission industry. A notable activated species in fusion systems is tritium and a limitation on tritium permeation into the coolant is one of the design requirements to be addressed in first wall and blanket designs.

RADIATION-HARDENED INSTRUMENTATION

Instrumentation on current heavily instrumented physics experiments can be broadly categorized as either diagnostics for measuring plasma parameters or instrumentation for vacuum, magnets and plasma heating systems. Beginning with TFTR (Tokamak Fusion Test Reactor, Princeton, N.J.) the next generation of fusion devices will operate with D-T plasmas and produce significant neutron radiation through the fusion reaction ($D + T \rightarrow He + n$). These devices will present new requirements for rad-hard instrumentation in control diagnostics, in test modules and for surveillance of component performance.

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