# US DEMO Blanket R&D: Status and Prospects

## 1. Overview
## 2. Solid Breeder Blanket

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US Home Team
Test Program Task Leader

<u>US Organizations Contributing to DEMO Blanket R&D</u>

ANL
UCLA
PNL
ORNL
INEL
LANL

First Meeting of the ITER Test Program Working Group;
Garching JWS, July 19-21, 1995

# US DEMO Blanket R&D Program

## DEMO Definition

- US DEMO definition was developed in the early 1980's. A present study (STARLITE) is evolving the definition, developing a conceptual design and examining pathways to DEMO

- DEMO goals and the range of major parameters available are adequate for the purpose of defining and guiding the blanket R&D program

## DEMO Blanket Design and Identification of R&D

- In the 1970's and 1980's, the US conducted comprehensive blanket studies to:

    1) develop and evaluate conceptual blanket (and reactor) designs

    2) identify the technical issues

    3) define R&D needs

    4) evaluate facility pathways

    Examples: BCSS, FINESSE, (also STARFIRE, DEMO,TPA, ARIES, STARLITE, etc.)

# Fusion Program Guidelines for DEMO Major Parameters and Features

| Parameter of Feature | European Union | Japan | Russian Federation | United States |
|---|---|---|---|---|
| Plasma Mode of Operation | Aim for steady-state; determine whether long-burn pulsed operation can be tolerated | Steady-state | Both pulsed and steady-state are being considered | Steady-state |
| Tritium Fuel Cycle: Global Tritium Breeding Ratio (TBR) | Self-sufficient<br><br>TBR > 1.0 | Self-sufficient<br><br>TBR > 1.0 | Self-sufficient<br><br>TBR: 1.05-1.1 | Self-sufficient<br><br>TBR > 1.0+ addition for doubling time |
| Power Output | Significant amounts of electricity | 3 GW fusion power | < 1.5 GW electric | Hundreds of MW electric |
| Neutron Wall Loading in $MW/m^2$ | 2-3 | up to 5.0 | 2-3 | 2-3 average<br>3-4 peak |
| Availability | Depends on DEMO mission; could be > 50% for reactor island | 70% | > 60 % | 50% net plant goal, but may start at 25% |
| Thermal Efficiency | Unspecified | 30-40% net | > 40% | > 30% net |
| Blanket Lifetime Neutron Fluence Goal in MW-$yr/m^2$ at First Wall | Depends on specific DEMO goals; could be 5 for first blanket and > 10 long-term | up to 7 | 15-20 | 10-20 |
| Environmental Consideration | Due account of environmental constraints | Low activation materials | Low activation materials | Low activation materials |

# US Perspective on DEMO Definition

## DEMO Mission/Goals*

| MISSION |
|---|
| The size, operation, and performance of DEMO must be sufficient to demonstrate that there are no open questions about the fuel cycle, safety, environmental impact, and economics of the first commercial fusion power plant. |

### GOALS

Demonstrate, through actual sustained operation of a fully integrated power plant system, that electrical power generation with fusion energy is:

- renewable (i.e. fuel cycle can be closed with doubling time suitable for fusion power economy)
- safe and licensable
- of low environmental impact
- economically competitive (i.e., competitive cost-of-electricity based on considerations of capital, operation/maintenance, and decommissioning costs and plant availability)
- reliable and maintainable

* These guidelines are suggested by the White Paper for use in DEMO-related studies until STARLITE has completed its work on the subject.

# US Perspective on DEMO Definition

## DEMO Major Parameters/Features*

<table>
<tr><th></th><th colspan="2">Value</th></tr>
<tr><td>Tritium Fuel Cycle</td><td colspan="2">self-sufficient (with 5-10 year doubling time)</td></tr>
<tr><td>Plasma Operation</td><td colspan="2">steady-state</td></tr>
<tr><td>Net Plant Availability (must be demonstrated for periods exceeding years)</td><td>> 50%</td><td rowspan="2">A scenario in which net plant availability during initial period of operation is 25% (power core availability of 30%) may be acceptable provided that net plant availability increases in later years of operation to >50% (power core availability of > 60%), which is sustained over many years of operation</td></tr>
<tr><td>Power Core Availability (assuming BOP Availability of 85%)</td><td>> 60%</td></tr>
<tr><td>Net Electrical Power Output</td><td colspan="2">hundreds of MWe</td></tr>
<tr><td>Net Thermal Conversion Efficiency</td><td colspan="2">> 30%</td></tr>
<tr><td>Overall Plant Lifetime</td><td colspan="2">30 years</td></tr>
</table>

*These guidelines are suggested by the White Paper for use in DEMO-related studies until STARLITE has completed its work on the subject.

# US Perspective on DEMO Definition

## DEMO Blanket System Performance Requirements

| | | Value | | |
|---|---|---|---|---|
| Neutral Wall Loading | Average<br>Peak | 2 - 3 $MW/m^2$<br>3 - 4 $MW/m^2$ | | |
| Surface Heat Flux | Average<br>Peak | 0.5 - 0.6 $MW/m^2$<br>0.75 - 1.0 $MW/m^2$ | | |
| Blanket System and Module Availability as Function of Power Core Availability (assuming 80 modules in blanket system) | | Power Core Avail. | Blanket System Avail. | Blanket Module Avail. |
| | | 75%<br>59%<br>52%<br>37%<br>30% | > 99.0%<br>97.6%<br>80.0%<br>50.0%<br>37.4% | > 99.98%<br>99.97%<br>99.69%<br>98.76%<br>97.90% |
| Blanket Module Mean Time Between Failure ($MTBF_M$) as Function of Mean Time to Replace (MTTR) for Power Core Availability of A = 30% and 60% (assuming 80 modules in blanket system and $MTBF_{BS}$ = $MTBF_M$/80) | | MTTR | $MTBF_M$ in FPY | |
| | | | A= 60% | A = 30% |
| | | 1 week<br>2 weeks<br>1 month<br>2 months | 62<br>125<br>271<br>542 | 0.92<br>1.84<br>3.98<br>7.96 |
| Blanket Module Lifetime | | 10 - 20 $MW\text{-}yr/m^2$ | | |
| Non-Vulnerable Tritium Inventory | | < 500 g | | |
| Decay Heat at Shutdown as Percentage of Full Operating Power | | < 0.4 % | | |
| Radioactivity per Watt Thermal Power | | < 1 Curie | | |
| Long-Term Radioactive Waste Characteristics | | recyclable within 100 years (surface gamma dose 100 years after shutdown < 25 microSv/hr) | | |

# Summary of Critical R&D Issues for Fusion Nuclear Technology

1. D-T fuel cycle **self sufficiency**
2. **Thermomechanical** loadings and response of blanket components under normal and off-normal operation
3. Materials **compatibility**
4. Identification and characterization of **failure modes, effects, and rates**
5. Effect of imperfections in electric (MHD) **insulators** in self cooled liquid metal blanket under thermal/mechanical/electrical/nuclear loading
6. **Tritium inventory** and recovery in the solid breeder under actual operating conditions
7. **Tritium permeation** and inventory in the structure
8. Radiation Shielding: accuracy of prediction and quantification of radiation production requirements
9. Plasma-facing component thermomechanical response and lifetime
10. **Lifetime** of first wall and blanket components
11. Remote maintenance with acceptable machine shutdown time.

# Key Fusion Environmental Conditions For Testing Fusion Nuclear Components

- **Neutrons** (fluence, spectrum, gradient)
  - Radiation Effects
    (at relevant temperatures, stresses, loading conditions)
  - Bulk Heating
  - Tritium Production
  - Activation
- **Heat Sources** (magnetic gradient)
  - Bulk (from neutrons)
  - Surface
- **Particle Flux**
- **Magnetic Field**
  - Steady Field
  - Time-Varying Field
- **Mechanical Forces**
  - Normal
  - Off-Normal
- **Thermal/Chemical/Mechanical/Electrical Interactions**
- **Synergistic Effects**
  - Combined environmental loading conditions
  - Interactions among physical elements of components

# US DEMO Blankets

- The US DEMO blanket R&D is now focused on two blankets:

  1) Helium-cooled Solid Breeder
     $He/Be/Li_2TiO_3/SiC$ (FS)

     Alternative Breeder: $Li_2O$, $Li_2ZrO_3$

  2) Self-cooled Li/V with insulator coating.

- The US recognizes that pressurized water may be a possibility for near-term, small-size systems where space and shielding are key issues.

- US studies have concluded that selection between solid breeder and liquid metal blankets can not be made prior to testing in fusion facilities.

- The US is very interested in developing blankets that satisfy tritium self sufficiency and efficient energy conversion conditions with attractive safety and environmental features. The US is motivated to seriously consider and advance low activation materials. However, the US also recognizes the difficulties in developing such low activation materials on the DEMO time scale (particularly if no other party is supporting such development). Initial testing in ITER during BPP will provide critical information on the realistic path to DEMO.

# DEMO Blanket R&D
# Current US Activities

1. Design Improvement
   Solid Breeders (UCLA), Liquid Metals (ANL)

2. Tritium Fuel Cycle Dynamic Modeling, Tritium Processing, Experiments, and Analysis
   UCLA, LANL, ANL

3. Neutronics R&D
   UCLA, LANL, ANL, UW, TSI

4. Solid Breeder Blanket R&D
   UCLA, ANL, PNL

5. Liquid Metal Blanket R&D
   ANL, UCLA

6. Structural Material R&D
   ORNL, ANL, PNL, UCLA

7. Blanket Safety [part of Safety Program]
   INEL, UW, UCLA

# DESIGN OF A He-COOLED SOLID BREEDER FOR THE U.S. REFERENCE DEMO BLANKET

- UCLA is currently leading the effort of a detailed reference solid breeder blanket design for DEMO

- DEMO Major Parameters used for the Reference Blanket Design are similar to those used by EU:

| | |
|---|---|
| Major radius | 6.3 m |
| Minor radius | 1.8 m |
| Aspect ratio | 3.45 |
| Plasma current | 20 MA |
| Fusion power | 2,200 MW |
| Average neutron wall load | 2.2 $MW/m^2$ |
| Average/Peak surface heat load | 0.4/0.5 $MW/m^2$ |
| Toroidal field on axis | 6 T |
| Mode of operations | Steady State |
| Impurity control | Double-Null Divertor |
| First wall protection | None |
| Number of TF-Coils | 16 |

# U.S. REFERENCE DEMO BLANKET
# FOR THE He-COOLED SOLID BREEDER

Major Features of the Reference SB-Blanket design:

- Breeder-Out-Tube (BOT) concept
- Both, the solid breeder and the Be are in the form of sphere-packed beds
- First wall and blanket are all poloidally cooled
- Solid breeder and Be regions are separated by coolant-panels
- No manifold/space behind the Inboard blanket
- All inboard manifolds and coolant supply lines are only at the top and/or bottom of the tokamak
- The Inboard blanket system consists of 32 segments; with a once-through flow path from bottom to top
- Each of the 48 Outboard segments are divided into one upper and one lower blanket module
- The shield is separate from the blanket, but can be incorporated readily as an integral part of the blanket module
- The first wall is an integral part of the blanket module
- The first wall and the blanket coolant panels use one manifold

# U.S. REFERENCE DEMO BLANKET FOR THE He-COOLED SOLID BREEDER

- DESIGN Features:

| | |
|---|---|
| Breeder material primary choice: | $Li_2TiO_3$ ($Li_2O$ is backup option) |
| Structural material: | Ferritic Steel |
| Multiplier material: | Beryllium |
| Breeder/Be form: | Sphere-packed |
| Coolant: | Helium |
| Purge gas: | Helium |
| First wall/Blanket | Integrated |
| Blanket/Shield | Integration is Optional |
| Coolant flow path | Poloidal |
| Inboard manifold location | Top and bottom of device only |
| Segment angle | 7.5° |
| Outboard blanket thickness | 50 cm |
| Inboard blanket thickness | 30 cm |
| Number of OB segments | 96 (48 upper and 48 lower) |
| Number of IB segments | 32 (optional: 64 - upper/lower) |
| Height of IB module | 7.2 m (optional 3.6m) |
| Height of OB module | 3.47 m (arc-length is 3.92m) |
| Height of IB shield | 9.5 m |
| Height of OB shield | 4.1 m (arc-length is 4.57 m) |

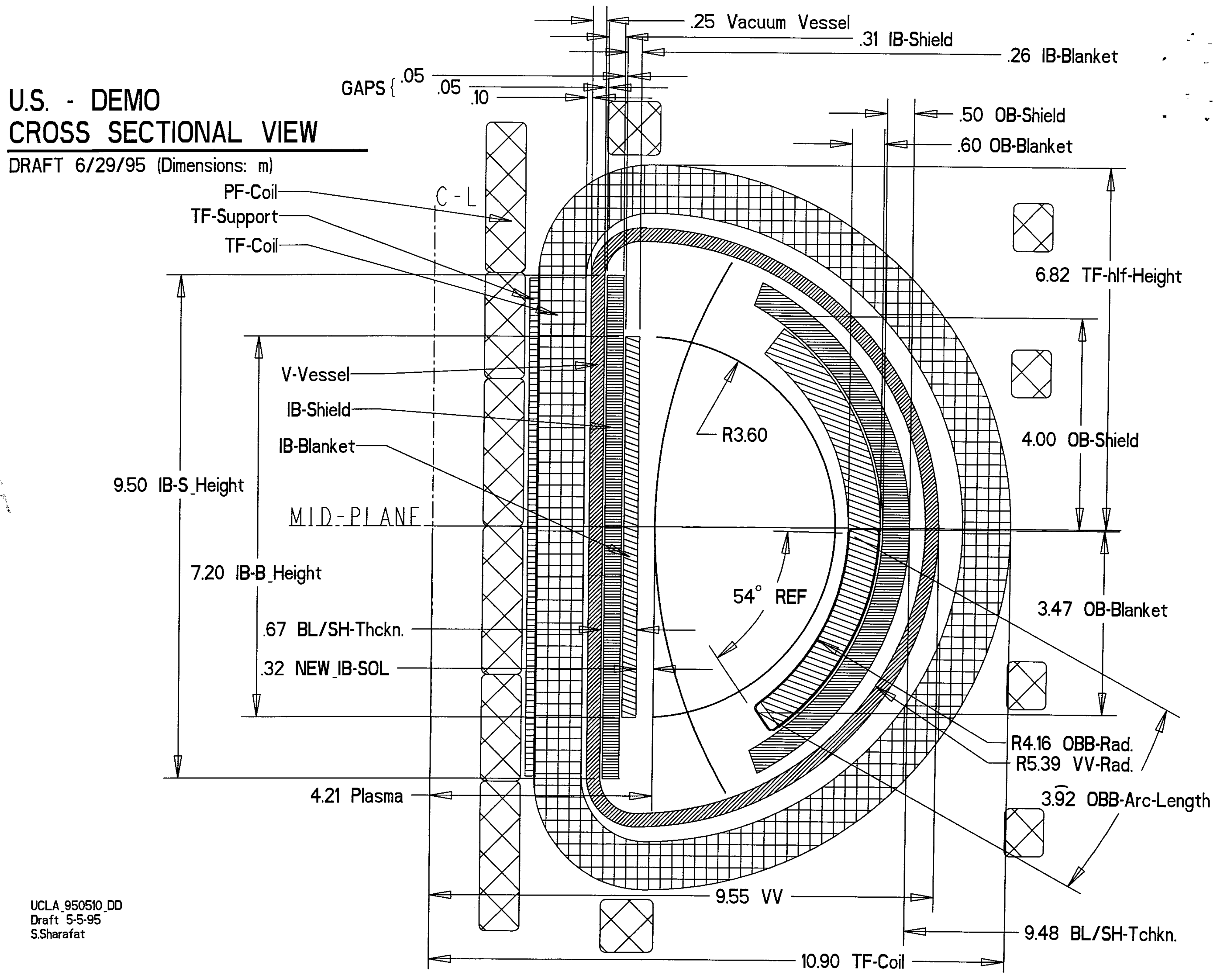
- Configuration of and design of the blanket module is being iterated with neutronic and thermalhydraulics calculations.

- Final Design Configuration has not been completed.

# U.S. REFERENCE DEMO BLANKET
# FOR THE He-COOLED SOLID BREEDER

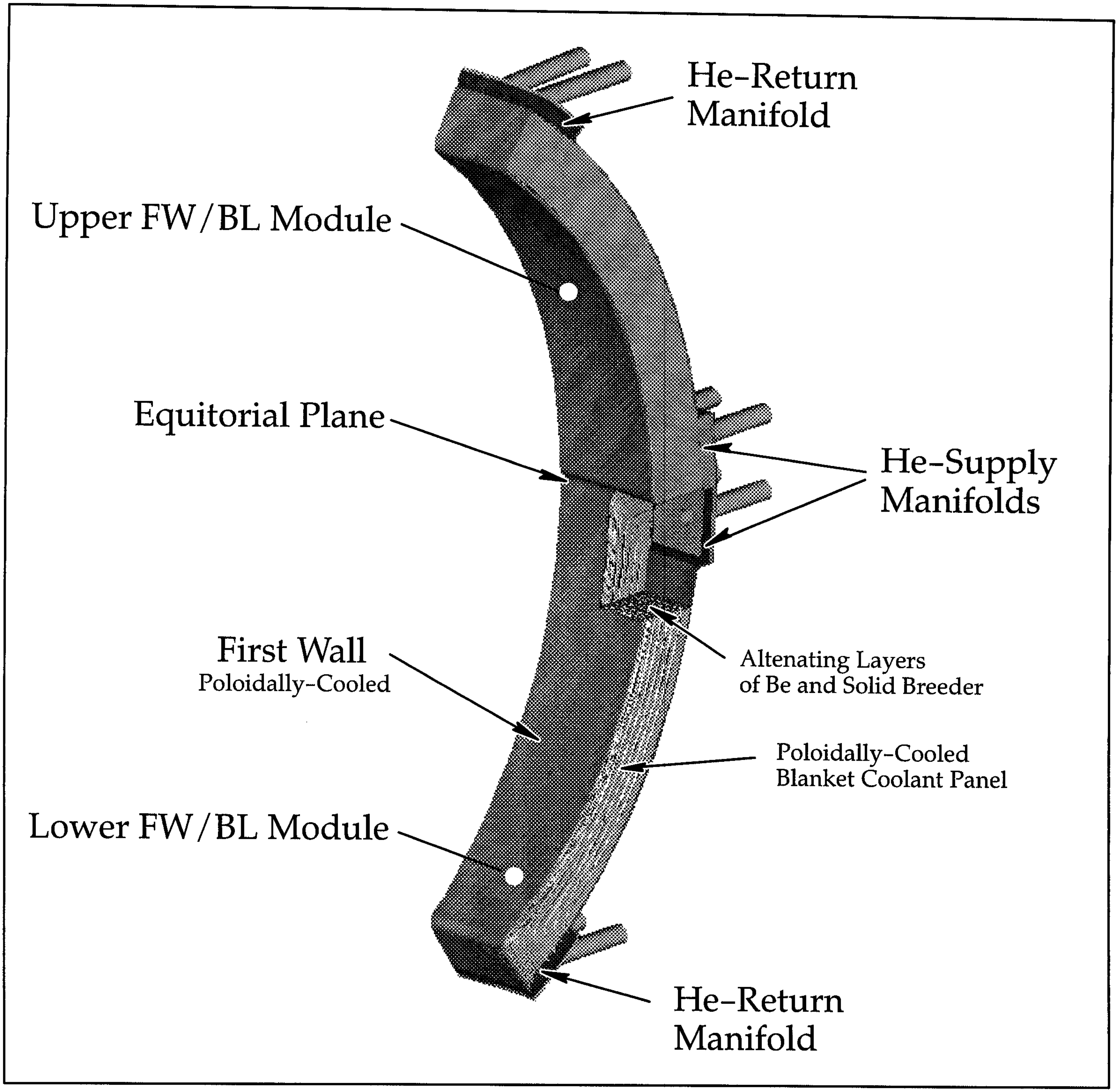
Preliminary Design Parameters:

| | |
|---|---|
| Breeding and Multiplier | Separated beds of $Li_2TiO_3$ and Be pebbles |
| Li6 enrichment (at.%) | 40 |
| Total blanket power | 2500 MW |
| Coolant Helium Pressure: | 10 MPa |
| Coolant Helium Pressure Drop | 0.47 MPa |
| Coolant Temperature:<br>Min.<br>Max. | <br>250°C<br>420°C |
| OB First Wall Temperature<br>Min.<br>Max. | <br>408°C<br>543°C |
| Max. Be Temperature | 660°C |
| Breeder Temperature<br>Min.<br>Max. | <br>320°C<br>930°C |
| 1-D TBR without ports | 1.47 |
| Tritium Purge System Pressure | 0.1 MPa |
| OB-FWCoolant velocity | 32 m/s |


U.S. - DEMO
CROSS SECTIONAL VIEW
DRAFT 6/29/95 (Dimensions: m)
.25 Vacuum Vessel
.31 IB-Shield
.26 IB-Blanket
GAPS { .05
.05
.10
.50 OB-Shield
.60 OB-Blanket
C - L
PF-Coil
TF-Support
TF-Coil
6.82 TF-hlf-Height
V-Vessel
IB-Shield
IB-Blanket
R3.60
4.00 OB-Shield
9.50 IB-S_Height
MID-PLANE
7.20 IB-B_Height
54° REF
3.47 OB-Blanket
.67 BL/SH-Thckn.
.32 NEW_IB-SOL
R4.16 OBB-Rad.
R5.39 VV-Rad.
4.21 Plasma
3.92 OBB-Arc-Length
9.55 VV
9.48 BL/SH-Tchkn.
10.90 TF-Coil
UCLA_950510_DD
Draft 5-5-95
S.Sharafat

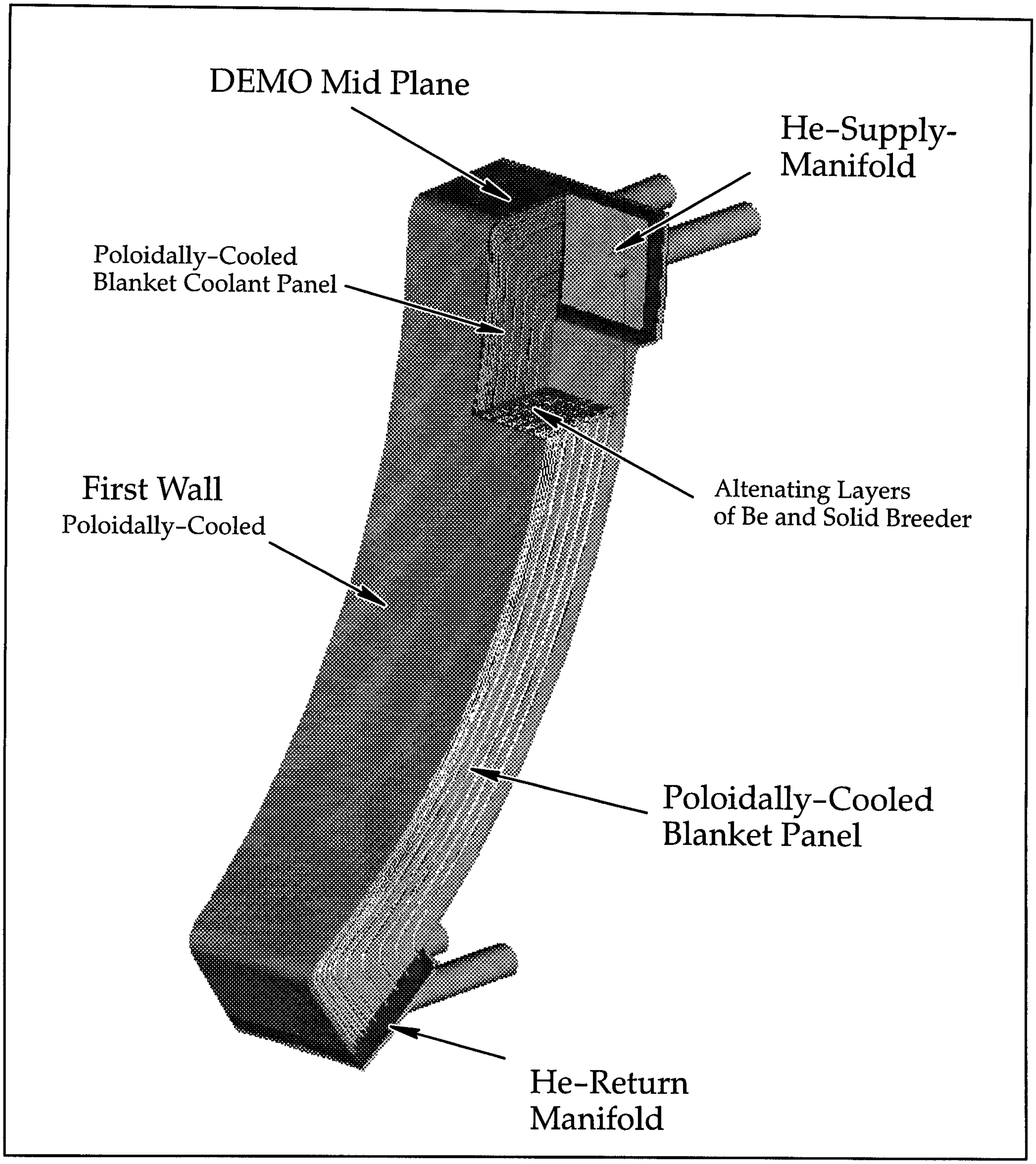
# U.S. DEMO He-Cooled Solid Breeder Reference Blanket

## One Complete First-Wall/Blanket Segment



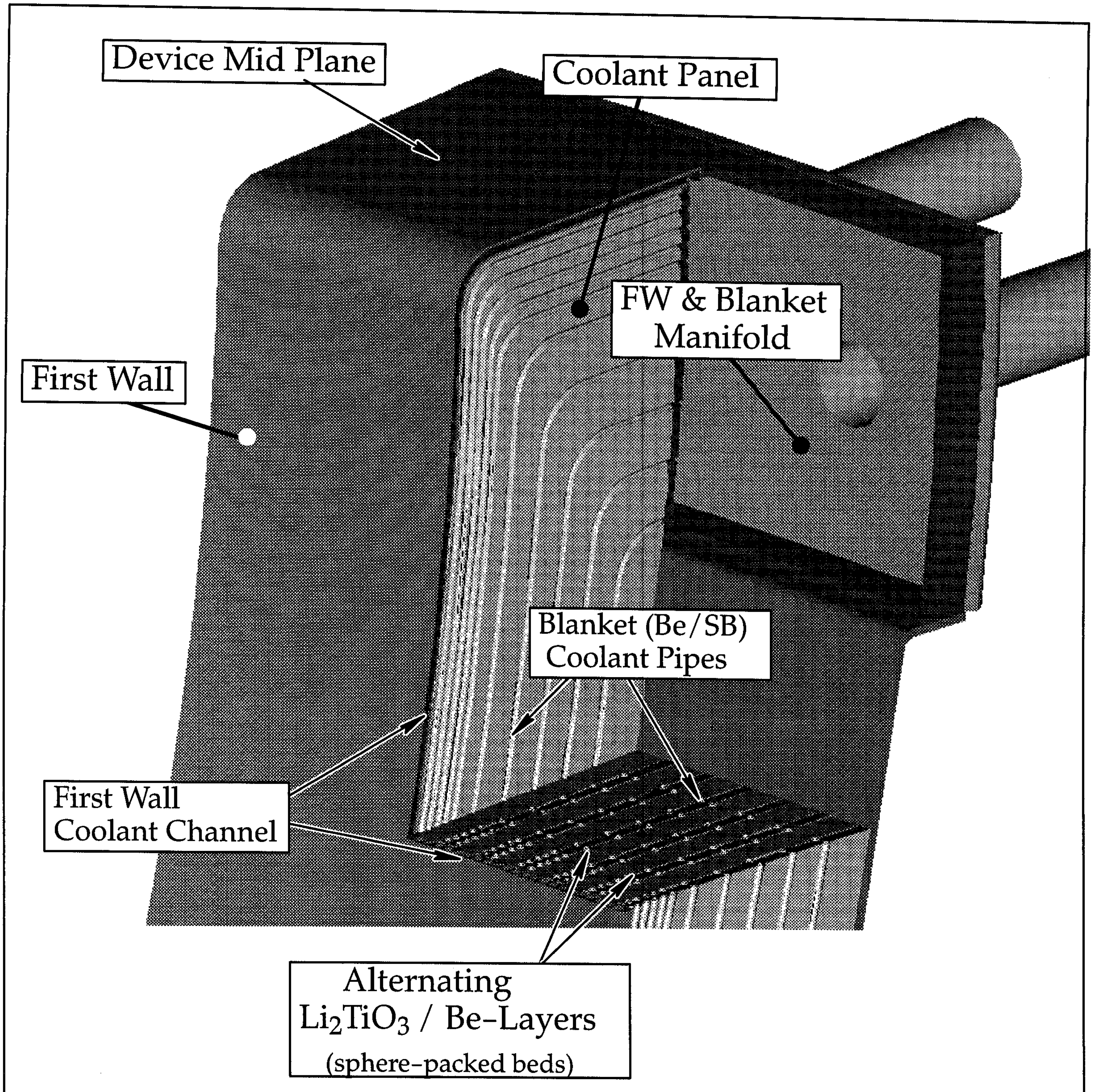
# U.S. DEMO He-Cooled Solid Breeder Reference Blanket

## Lower First-Wall/Blanket Segment



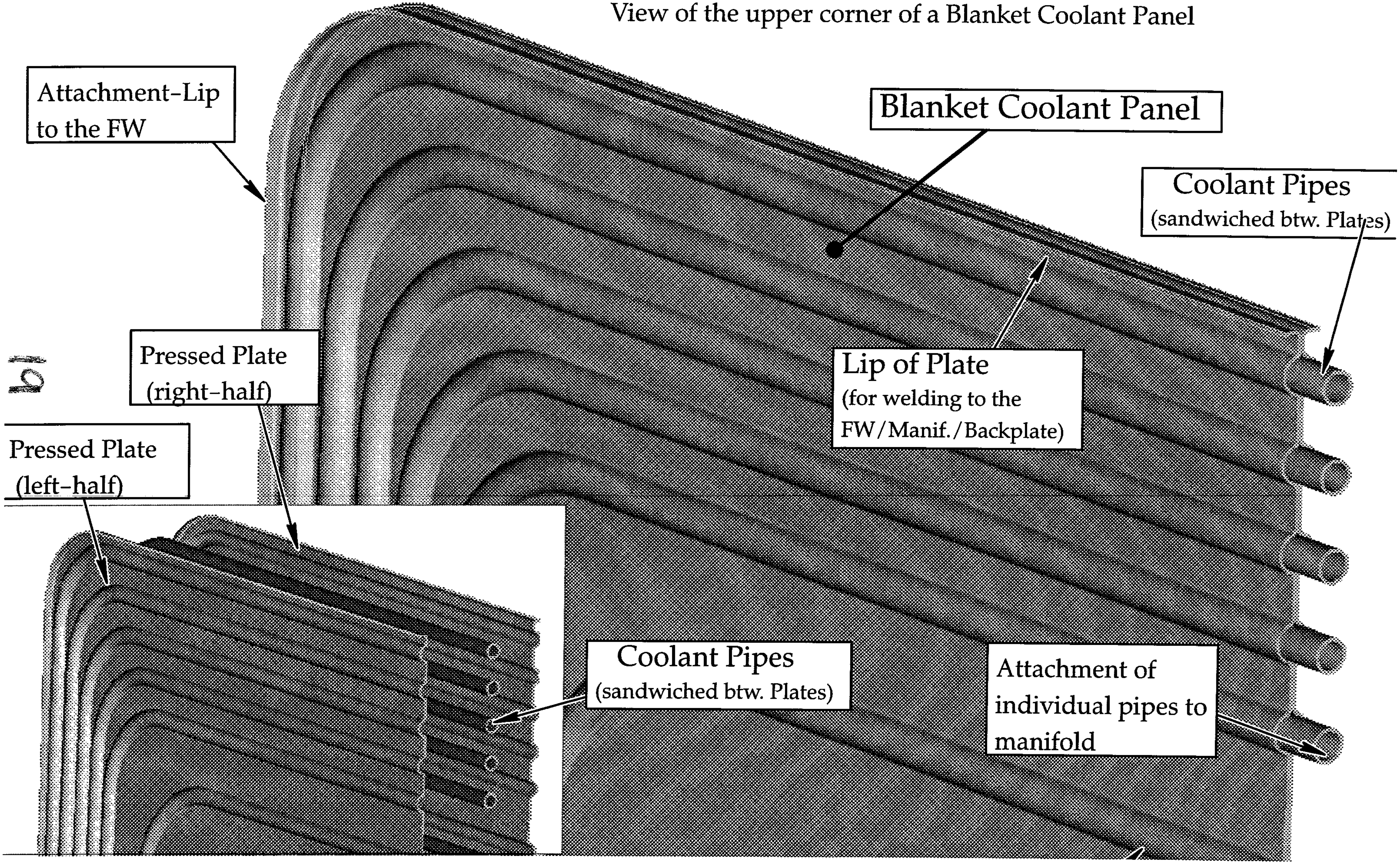
# U.S. DEMO He-Cooled Solid Breeder Reference Blanket

## Cutout View of the First Wall/Blanket/Upper Manifold



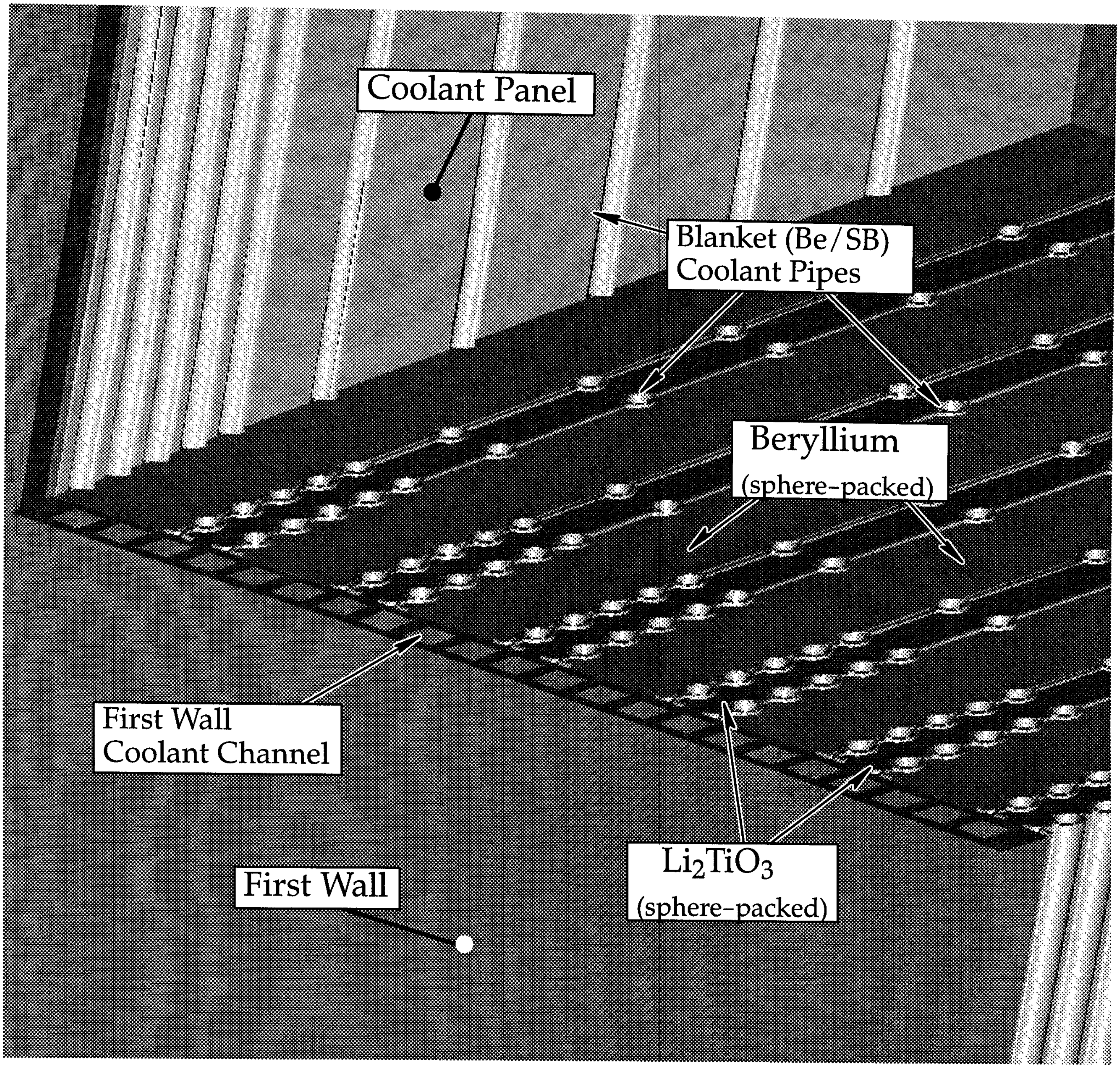
# U.S. DEMO He-Cooled Solid Breeder Reference Blanket

View of the upper corner of a Blanket Coolant Panel



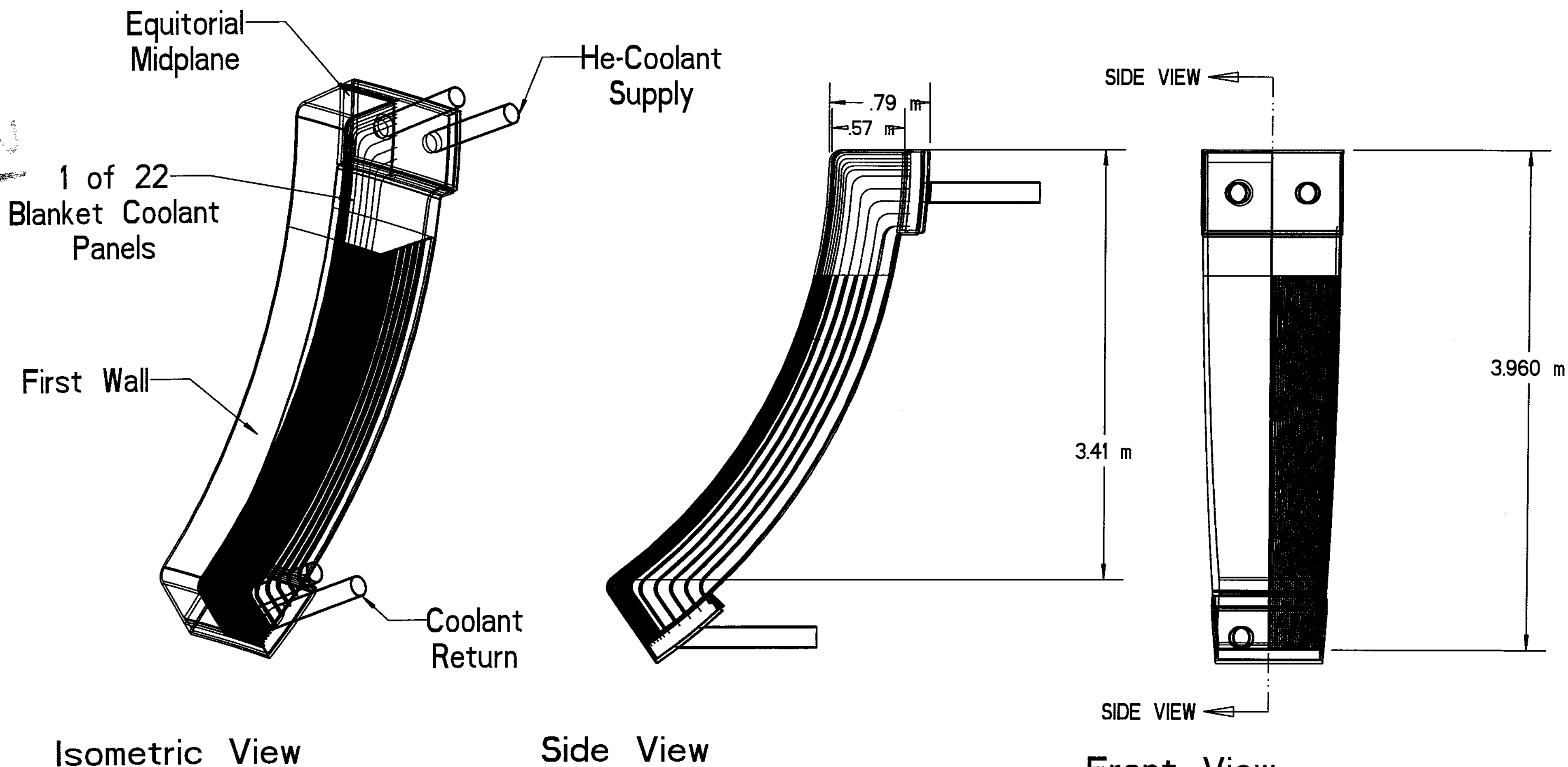
# U.S. DEMO He-Cooled Solid Breeder Reference Blanket

## View of a Horizontal Cut Through the FW/Blanket



# U.S. Reference DEMO Blanket
# For the He-Cooled Solid Breeder

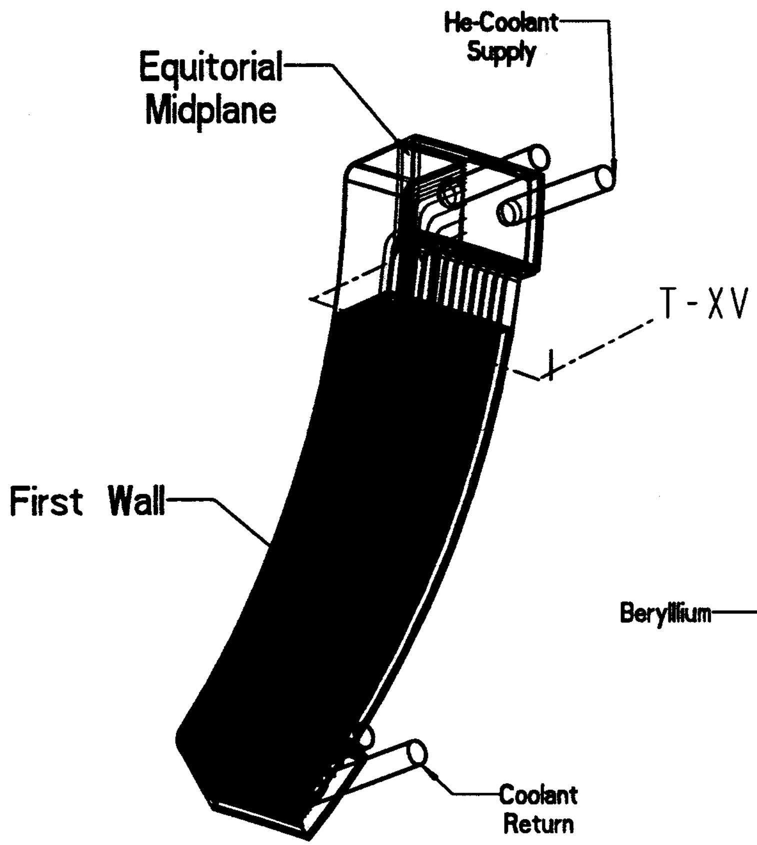
## Lower Outboard First Wall/Blanket Module




UCLA: Draft 6/29/95
UCLA: S. Sharafat

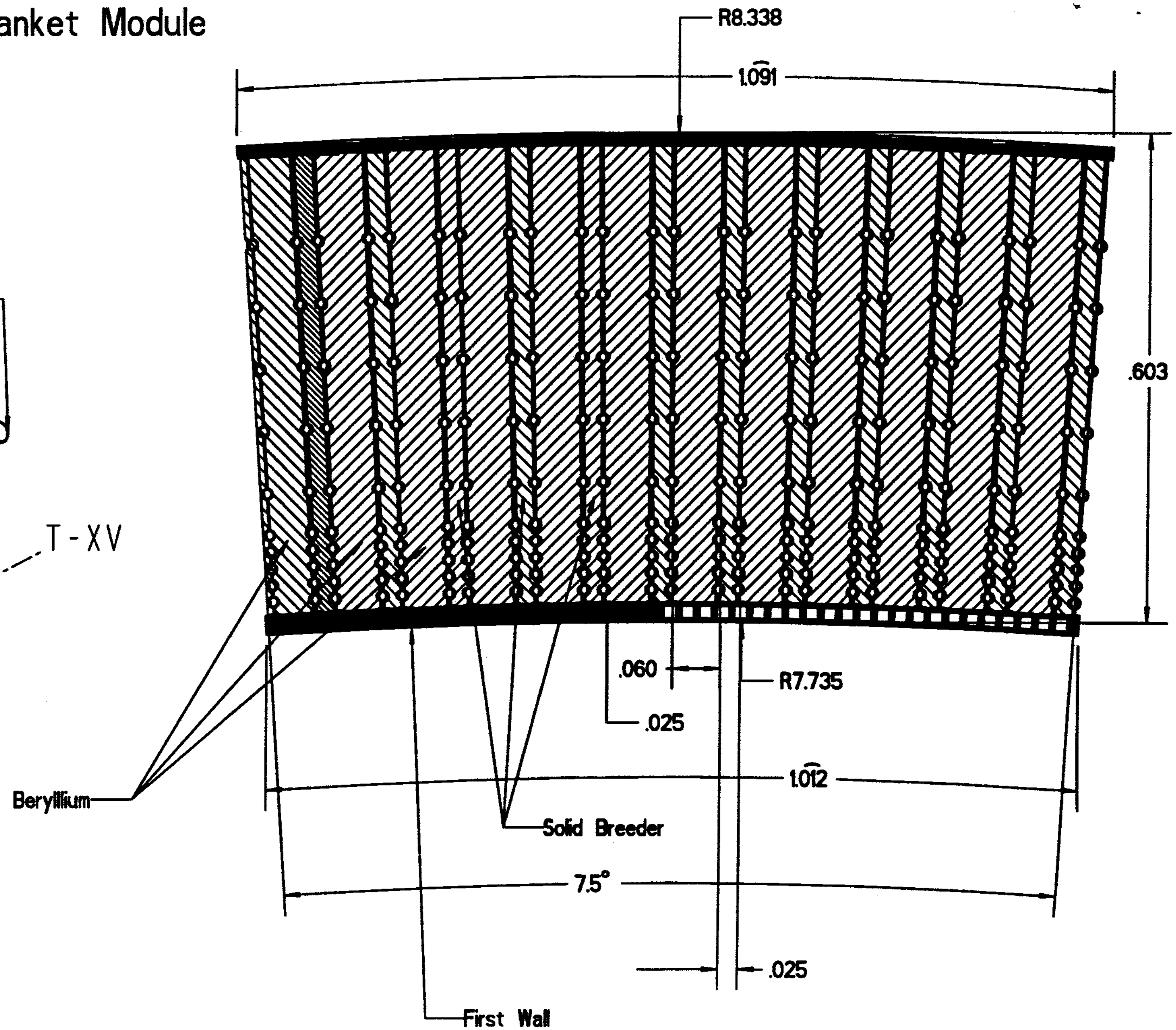
# U.S. Reference DEMO Blanket
# For the He-Cooled Solid Breeder

## Lower Outboard First Wall/Blanket Module




Isometric View

Toroidal Cross Section



BEATRIX-II In Situ Tritium Recovery Irradiation

- IEA collaborations (Japan, Canada, U.S.)
- Fully instrumented, in-situ tritium recovery tests in FFTF
- $Li_2O$ Rings and Sintered Pellets, $Li_2ZrO_3$ Spheres (1.2 mm), and Be
- 450 to 1100°C temperatures
- 4.5% Burnup in fast neutron spectra
- PIE completed/Data reduction in progress

BEATRIX-II Conclusions — $Li_2O$

- Much more stable than previously thought even up to 900°C
- Limited mass transport
- Low tritium inventories (<1wppm) maintained at higher burnups
- Restructuring of $Li_2O$ microstructure at high temperatures
- Temperature stability

BEATRIX-II Conclusions — $Li_2ZrO_3$:

- Very stable up to 1100°C
- Low tritium inventories
- Temperature stability

FUBR-1B Closed Capsule (BEATRIX-I) Experiment:

- $Li_2O$, $Li_2ZrO_3$, $Li_2O$, $Li_4SiO_4$, and $LiAlO_2$
- Up to 12% burnup (5 years)
- Up to 1200°C
- Irradiation & neutron radiography completed
- PIE in progress

FUBR-1B Initial Conclusions:

- Limited swelling for ternary ceramics (<2%)
- $Li_2O$ swelling higher but does not deform capsules
- Reasonable structural integrity with limited cracking from thermal stresses

## Ceramic Breeder Materials Thermodynamics

- When moisture comes in contact with lithium ceramics, lithium hydroxie will form. Understanding the consequences of this reaction is the only approach to defining the operating regime of a breeder blanket.

- Experimental investigation of the solubility of $H_2O$ in ternary ceramics ($Li_2ZrO_3$ and $Li_2TiO_3$) is in progress

- Ab-initio quantum mechancial calculations have been used to determine the vapor pressure of lithium hydroxide at low temperatures (<400°C).

- Complementary calculations are in progress to assess the compatibility between the lithium ceramics and low activation structural materials.

## Tritium Inventory & Modeling in the U.S.

- Substantial tritium modeling activities are ongoing at UCLA. There is a strong collaboration between UCLA and ANL

- In-Situ Tritium Recovery Data (ANL) and post irradiation measurements (PNL) and laboratory experiments are used to verify tritium release models.

- The tritium model codes were also made available to other parties as part of international collaborative activities.

## Tritium Inventory & Modeling in the U.S.

Tritium Release Models (UCLA):

CERAMIC BREEDERS:

MISTRAL: A comprehensive model for T-transport in ceramic breeders has been developed and used for DEMO Blanket (ITER blanket) and analysis

The model includes:

- Surface
- Bulk diffusion
- Solubility

SINGLE CRYSTALS:

MISTRAL-SC has been developed specifically for modeling tritium transport in ceramic breeder single crystals

- Focus is on better understanding fundamental processes and characterization of property data bases

BERYLLIUM:

- Comprehensive model has been developed for better understanding of tritium behavior in Be

Models and codes are verified against irradiation and laboratory experiments.

## Thermal Control Issues for Solid Breeder Blankets

- Maintain Breeder within allowable temperature limits
  — uncertainties due to manufacturing and operating conditions
  — accomodate power variations
  — correct for radiation-inudced changes in behavior

- Generic issues for blankets:
  — Critical issues for ITER
    - Breeding Blanket
    - Test Modules
  — DEMO and commercial power reactors

# GOALS

## Thermomechanics of Solid Breeder Blankets R&D in the US

- Develop innovative design solutions to thermal control problem e.g. **active control mechanisms**

- Develop modeling capability to accurately predict blanket behavior

- Generate empirical data used for design

- Validated concepts through experiments and modeling

Activities on

Thermomechanics of Solid Breeder Blankets R&D in the US

**Particle Bed Heat Transfer Experiments and Modeling**

- Packing experiements
- Effective thermal conductivity
- Wall conductance
- Purge flow characteristics

**Contact Resistance Between Metal Surfaces (sintered Be and Steel)**

- Studies of effect of surface conditions, contact pressure, thermal deformations, background gas pressure and consumption

**Mechanical Interactions in Pebble Beds**

- Effective coefficient of thermal expansion
- Effect of internal (thermal) expansion coefficient $k_{eff}$ and h
- Effect of external pressure/deformation on $k_{eff}$ and h

**Unit Cell Experiments**

- Study geometrical effect of elements of solid breeder blanketsInteraction between thermal & mechanical behavior of neighboring elements

## PARTICLE BED HEAT TRANSFER EXPERIMENTS AND MODELING

| Experiment | Description | Issues | Comments |
|---|---|---|---|
| **PBX** | Pebble Bed Heat Transfer Experiment at Low Temperature | • Gas Phase Control<br>• Effective Bulk Conductivity<br>• Wall Conductance<br>• Pressure Drop | Basic Properties |
| **ICE** | Interface Conductance Experiment | • Be With Surface Roughness<br>• High Contact Pressure<br>• Variable Heat Flux<br>• Control of Gas Phase | Basic Properties |
| **HTBX** | Pebble Bed Heat Transfer at High Temperature | • Mechanical Response to Thermal Expansion<br>• Effect of Mechanical Constraints on Heat Transfer | Basic Properties & Separate Effects |
| **HiTeC** | High Temperature Cyclic Heat Transfer in Prototypic Solid-Breeder Blanket Geometry | • Be or Ceramic Pebble Beds<br>• Independent Control of Temperature and Gradient<br>• Bulk Conductivity<br>• Wall Conductance<br>• Effect of Bed or Clad Deformation | Separate Effects |
| **UNICEX-B**<br>**UNICEX-S**<br>(under design) | Solid Breeder Blanket Unit Cell Experiment | • Thermomechanical Interactions<br>• Breeder & Multiplier at Prototypical Conditions<br>• Simulation of Bulk Heating<br>• He Coolant, He purge | Multiple Effects |

UNICEX-B: Packed Bed Experiments

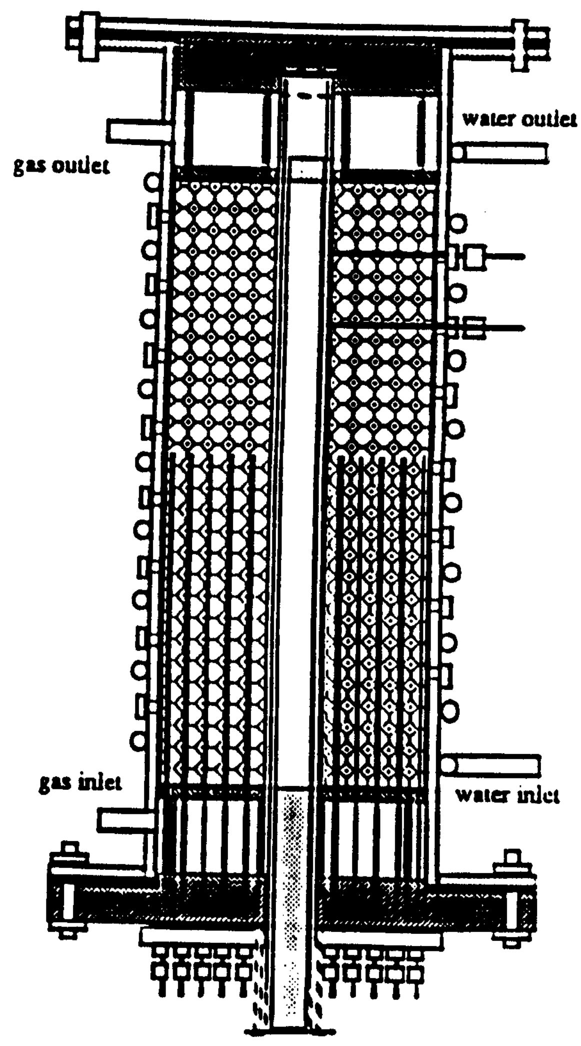
UNICEX-S: Sintered Block Experiments

# Thermomechanics of Solid Breeder Blankets: Experiments, Modeling, and Analysis (UCLA)

## METALLIC PARTICLE BED EXPERIMENT – PBX

Effective thermal conductivity was measured:

- for a range of He and $N_2$ gas pressures
- for several single-size and binary Al beds
  - different particle sizes
  - different surface roughness
- for a range of porosities
- reproducible in 2 different test sections




**The effective thermal conductivity of metallic packed beds shows substantial variation with gas pressure (and composition) - in some cases as much as a factor of 4. This clearly demonstrates the possibility of active temperature control in solid breeder blankets.**

- Data was provided to the U.S. ITER team, to CFFTP, JAERI, and in several national and international meetings, including ISFNT-2/3

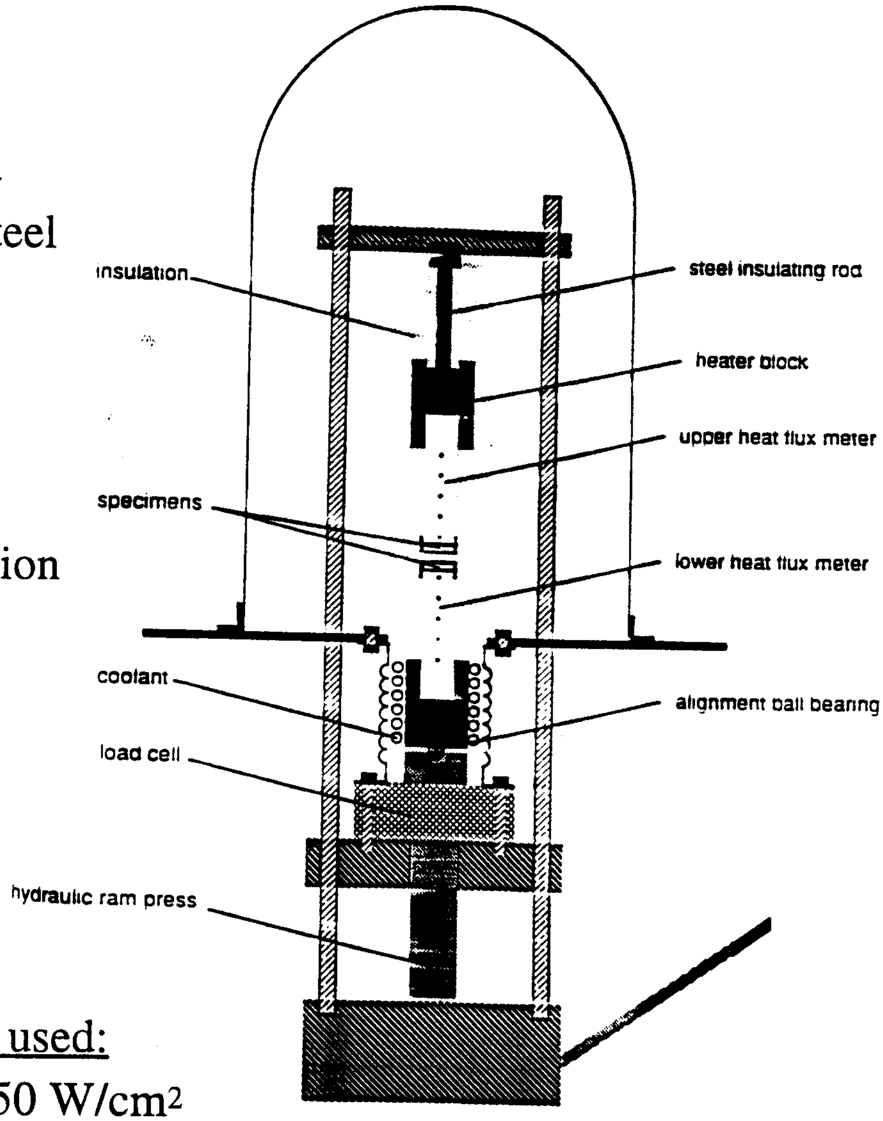
- JAERI has initiated studies of active control for application in their ITER blanket design.

Thermomechanics of Solid Breeder Blankets:
Experiments, Modeling, and Analysis (UCLA)

## INTERFACE CONDUCTANCE EXPERIMENT – ICE

Motivation:

- Existing models and data show large discrepancies, even for steel interfaces




EXPERIMENTAL GOALS:

- h has been measured as a function of:
  - contact pressure
  - surface roughness
  - cover gas pressure
  - heat flux

Prototypical values of parameters used:

| | | |
|---|---|---|
| Heat flux | q | 0 - 50 $W/cm^2$ |
| Surface roughness | $\partial$ | $1 \leq \partial \leq 50\ \mu m$ |
| Cover gas | | $N_2$, He |
| Cover gas pressure | p | 1 Torr $\leq p \leq$ 1 atm |
| Contact pressure | $P_c$ | 0 - 20 MPa |
| Contact conductance | h | $200 \leq h \leq 20{,}000\ W/m^2\text{-}K$ |

## INTERFACE CONDUCTANCE EXPERIMENT (ICE) cont.

- Our 304 SS data correlates well with Song data and Yovanovich model.

The contact conductance at interfaces depends on several variables:

- Contact Pressure
- Surface Roughness
- Bed material

Contact Conductance FINDINGS:

(1) Active control is possible over a limited range of parameters, and
(2) Rougher surfaces have a lower h, but larger stable range of operation

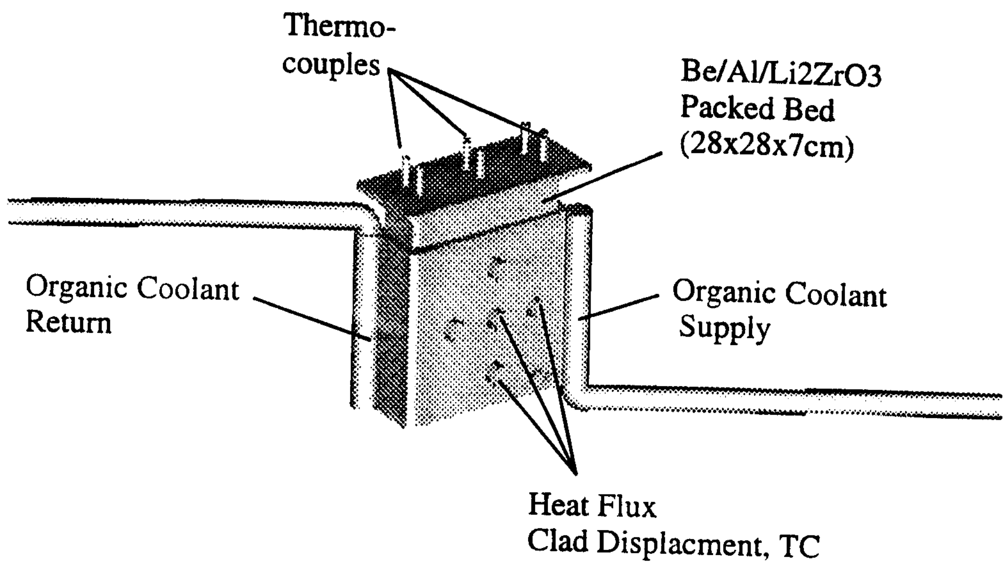
BED/CLAD Thermomechanical Interactions FINDINGS:

(1) Changes in contact area can result in a major change (factor of 2) in $k_{eff}$ and h

(2) Forces are amplified due to relatively small contact areas (stress amplification factor depends on rc/rs)

# HIGH TEMPERATURE CYCLIC EXPERIMENT IN PROTOTYPICAL BLANKET GEOMETRY

## — HiTeC —




## HiTeC OBJECTIVE:

HiTeC experiment is designed to help resolve the uncertainties associated with packed bed thermal and mechanical behavior, including the influence and interactions of the cladding under ITER-relevant environmental conditions

## HiTeC GOAL:

Measure the thermal and mechanical response of a set of geometrically idealized test articles which represent unit cells of the most attractive solid breeder candidates for ITER and DEMO reactors

## HiTeC Phase-II EXPERIMENTAL ACTIVTIES

HiTeC is being fitted with additional diagnostics to investigate the low h-values measured for prototypical geometries, compared with small sample results

- HiTeC-II can be outfitted to run measurements with Beryllium instead of Al.

- HiTeC-II will run at higher temperatures:

| | |
|---|---|
| ¬ Be (Al) pebble-bed | $T_{max}$ ~ 450°C |
| ¬ $Li_2ZrO_3$ pebble-bed | $T_{max}$ ~ 685°C |
| ¬ Bed T-gradient: | $\Delta T$ ~ 250°C Be/Al-bed |
| | $\Delta T$ ~ 350°C $Li_2ZrO_3$ |
| ¬ Coolant | $T_{max}$ ~ 300°C |

- Cyclic effects to be investigated after completion of steady-state measurements and modeling

Thermomechanics of Solid Breeder Blankets:
Experiments, Modeling, and Analysis (UCLA)

**UNICEX - Unit Cell Solid Breeder Thermomechancis Experiment**

PROGRAM OBJETIVES

- **Prototype demonstration**

  Demonstrate ability to maintain acceptable temperatures in a (multiple-layer) unit cell through both passive and active means:

<u>Active</u> control: varying the gas composition and pressure in the bed;
<u>Passive</u> control: over-heating results in thermal expansion, which leads to better contact conductance)

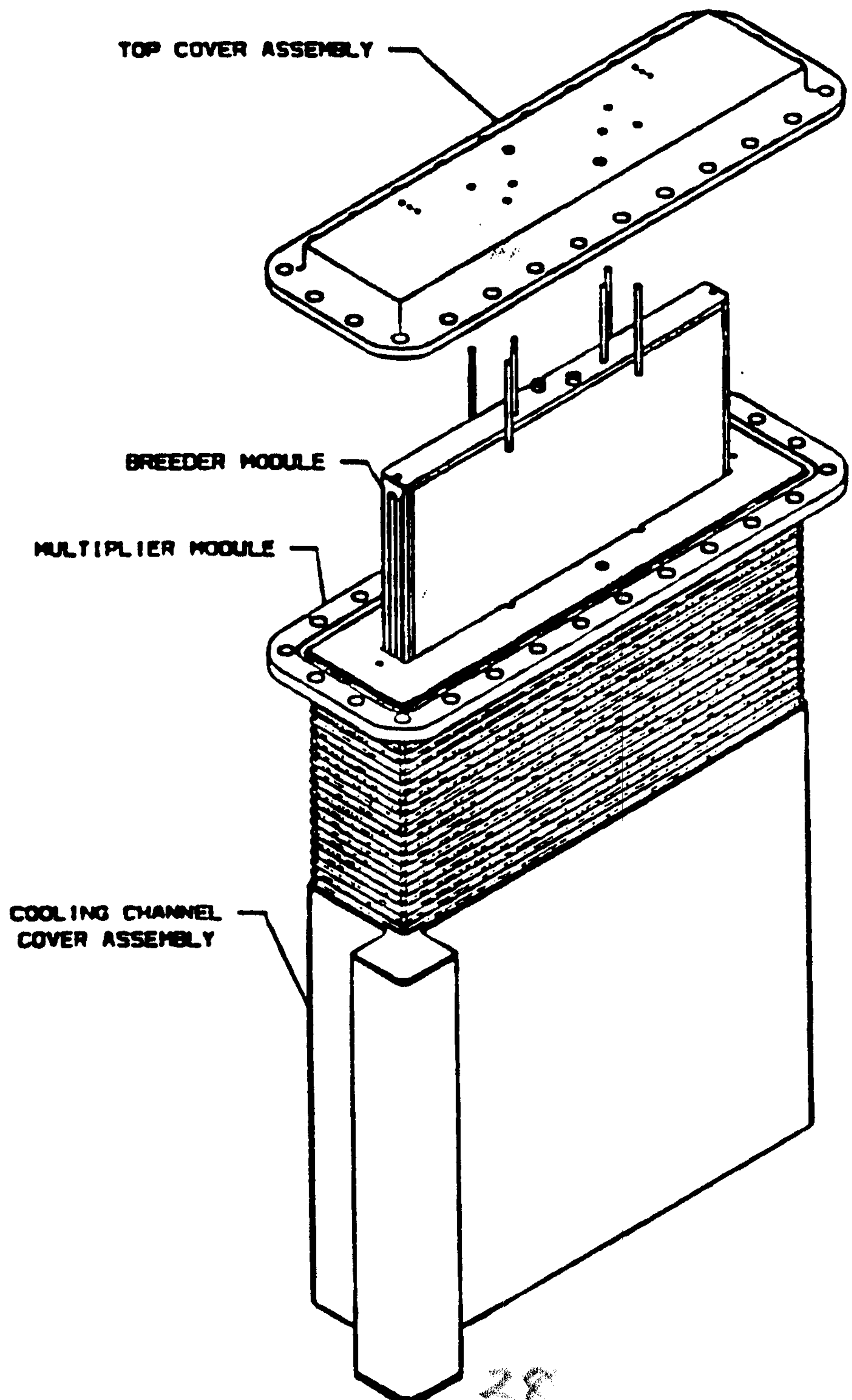
- **Design improvement**

  Develop blanket designs with good thermal control features, including the development of engineering design detail to uncover and resolve design-specific problems

- **Develop fusion-applicable experience**
  - Fabrication
  - Engineering design
  - Test operation
  - Experimental (measurement) techniques

# UNICEX TEST ARTICLE CONFIGURATION

- The core of UNICEX is the design, construction and testing of a prototypical (~28 x 28 x 7 cm) unit cell test articles.



# SiC/SiC COMPOSITES FOR FUSION APPLICATIONS

Advantages of SiC/SiC

- Low activation and afterheat
- "Engineerable" properties
- Fracture toughness
- High temperature properties
- Corrosion resistance
- Radiation stability of beta Sic

- Participating Laboratories, Industry and Universities

  General Atomics
  Oak Ridge National Lab.
  Pacific Northwest Lab.
  Rensselaer Polytechnic Inst.
  Univ. of California at Los Angeles

- International Collaborations and Activities

  DOE/Monbusho: Irradiation's and testing, creep
  ISPRA: active program
  France: program being developed
  IEA: workshop approved

# Status of SiC/SiC Composite Materials

Issues in application of SiC/SiC

- Radiation stability of SiC/SiC composite
  - — He/H/Na generation rates
  - — C transmutation-SiC stability
  - — Thermal conductivity: routes for improvement
- Hermetic behavior/pressure boundary
- Chemical compatibility with fusion environments
- Processing/fabrication/joining
- Qualification/design methodology
- $^{26}$Al and the low activation criteria
- Cost

- Feasibility phases
  - Design studies demonstrate potential
  - Investigations of basic properties
  - Identify critical or limiting properties
  - Investigations of chemistry, microstructure and property relationships

Completed for:
- Base material properties
- Hermetic properties
- Thermal conductivity
- Coolant compatibility
- Joining
- Fatigue/shock

Feasibility Phase I - completed
- Demonstrate a stable fiber at 800-1000°C, to 5-10 dpa, < 1% density change
- Completed based on KAPL data: 4/1/95

# Prospects of SiC/SiC Composite Materials

Feasibility Phase II (ongoing):

—Demonstrate stable composite behavior at 800-1000°C, 5-10 dpa, irradiated fracture strength and strain greater than 2/3 of unirradiated properties and greater than 200 MPa.
—Evaluate effects of transmutation products (He,Na) on dimensional stability and fracture strength.
—Demonstrate availability based on a response to an RFQ to supply SiC/SiC with desired fiber/interface/matrix.

- Materials Development Phases

Material development phase I:
Composite Fabrications,
Composite testing and test methodology development
Composite qualification development
Material development phase II:
- Extend phase I irradiation database and add more advanced fibers and interfaces as available
- Develop fabrication and joining technology
- Define chemical compatibility sufficient to support system design, experimental data to support design.
- Obtain materials data base on prime and back-up candidate materials

- Materials engineering phase: 2001

- ITER Test Module Development

+ Data base
+ Prototype test module
+ Materials qualification scheme

- Database for DEMO

Status Summary of SiC/SiC Composites For Fusion Applications

- SiC/SiC composites exhibit promising properties for fusion applications
- Other CFCC programs are major drivers in the development of improved fibers, interfaces, processing, and development of joining technology and a materials database.
- Limited radiation effects data base indicates that Nicalon fibers do not exhibit sufficient radiation resistance.
- Very limited data indicates that Hi-Nicalon has far superior radiation resistance than Nicalon.
- There are several fibers, interfaces and matrix processing routes being developed by other programs that look promising for fusion applications.
- Hermetic behavior, demonstrated radiation resistance of composite materials, assessment of transmutation effects (He, Na) are the primary feasibility issues.
- A program plan to assess the feasibility issues and materials development and engineering has been developed and is partially implemented.

IEA Program is in Progress

- Two 5-ton heats of IEA modified F82H (8Cr-2WVTa)have been produced
- A 1-ton heat of JLF-1 (9Cr-2WVTa) and three small heats that are variations on this composition have been produced
- A coordinated research program is in the planning stage; preliminary testing has begun on the new heats
- Objective of collaborative program: to demonstrate the feasibility of using a reduced-activation ferritic/martensitic steel for DEMO and beyond

Feasibility Issues:

- Low -temperature limits imposed by irradiation embrittlement
  - integration of design and material properties so that the applied loads as a function of displacements are always less than those needed to initiate unstable crack growth conditions
- Development of stable nitride (Li) or oxide (Pb-Li) insulating coatings
  - Mechanical integrity under cyclic loads
  - radiation-induced increases in conductivity
  - self-healing
- Corrosion resistance in water
- Post-weld heat treatment

## Data Required For Design Performance Analysis

| Out-of-Reactor Effects | Irradiation Effects |
| --- | --- |
| Tensile | Tensile |
| Fracture toughness | Swelling and Microstructure |
| Fatigue | Fatigue |
| Thermal Creep | Fracture Toughness |
| Charpy Impact | Charpy Impact |
| Thermal Aging | Irradiation Creep |
| Creep-Fatigue | Creep-Fatigue |
| Corrosion/Compatibility | Stress-Relaxation |
| Stress-Relaxation | |
| Fatigue Crack Growth | |
| Thermal Fatigue | |
| Multiaxial Fatigue | |
| Hydrogen Effects | |

# U.S. Test Program for IEA Collaboration to Emphasize Fracture Properties

- <u>Micromechanics of Fracture:</u> Mechanical test combined with confocal microscopy, fracture surface reconstruction and finite element modeling to develop micromechanical models of fracture and a toughness data base

- <u>Size Effects/Crack Geometry Effects/Advanced Fracture Mechanics:</u> Mechanical testing to determine effect of specimen geometry, specimen size, and crack geometries in fracture criteria

- <u>Microstructure-Hardening-Fracture Correlation:</u> Assess validity of fracture models with irradiation-damage microstructure

- <u>Other</u>
  - low-cycle fatigue
  - composition optimization
  - helium effects
  - low-cycle fatigue

# U.S. Neutronics R&D Highlights

- U.S. Neutronics R&D covers neutronics issues everywhere in the system:

  blanket, bulk shield, magnets, reactor building, biological shield

- Neutronics Parameters:

  - Neutron and Photon Transport
  - Tritium Breeding
  - Nuclear Heating
  - Induced Radioactivity
  - Decay Heat
  - Biological Dose
  - Transmutation

- R&D Activities:

  - Methods and Codes Development.
  - Data Measurements and Evaluation.
  - Measurement Techniques Development.
  - Integral Experiments and Analysis.
  - Safety Factors and Uncertainty Estimate Methodology Development.

# U.S. Neutronics R&D Highlights (Cont'd)

- U.S. Neutronics R&D activities have included many successful collaborative efforts with other countries
    - The most prominent is a very successful 10-year collaborative program with Japan.
    - Other collaborative activities with Europe, Russia, and PRC.
- Experimental Facilities:
    - The U.S. uses neutron source facilities in other countries.
    - TFTR D-T shots have been used to perform measurements on radioactivity and shielding.
    - A small laboratory for measurement technique development is maintained at UCLA.
- U.S. Neutronics activities are structured to optimally serve the many interrelated needs of ITER and DEMO.

# <u>Integral Experiments and Analyses</u>

## (1) <u>U.S.-JAERI</u>

A decade-long collaborative effort with JAERI on verification of tritium production rate in *tens* of $Li_2O$ breeding blankets with various design features (FW, coolant channels, Beryllium multiplier, Large penetrations, etc). The Fusion Neutronics Source (FNS) 14 MeV facility at JAERI was used. State-of-the-art design codes/data bases were used (e.g. Monte Carlo code, MCNP, ENDF/B-V data).

* Experiments and analysis were performed for:

- Tritium Production Rate (TPR) for *verification of meeting Tritium self-sufficiency condition*

- Total Nuclear Heating Deposition Rate in various FW/Blanket materials.

- Radioactivity/Decay Heat in various FW/Blanket/shield/PFC materials.

* A special issue of Fusion Technology in August and September will have 20 papers from the U.S. and Japan which describe this major effort

* Additional effort is underway to perform further analysis of the huge experimental and analytical data generated in the collaboration.

(2) **IEA Collaboration:**

Plans are underway to initiate IEA collaboration to perform experiments and analysis on other breeding materials (e.g. Li2TiO3, Li2ZrO3, and LiAlO2, ).

(3) **TFTR Experiments:**

Neutronics experiments were conducted on TFTR during recent D-T shots. They cover radioactivity and shielding. Analysis is underway. Experiments on long-lived isotopes are planned for future D-T shots.

# Experimental Techniques Development

## (A) Recent Developments

- Proton Recoil Counters (PRC) for neutron flux measurements in the range of few keV-2 MeV.

- Li-foil techniques for TPR measurements by scintillation.

- Sensitive calorimeters (probes) for measuring total nuclear heating (temperature changes as low as $\sim 10^{-5}$ K/s were detected)

## (B) Future Effort

- Further measuring techniques' development is needed for the fusion environment of ITER and DEMO (e.g. magnetic field, high temperature, etc) for the following items:

  - *Neutron/gamma spectrum*
  - *TPR*
  - *Nuclear Heating and Radioactivity*
  - *post shutdown decay heat*

# Computational Tools Development and Nuclear Data Generation/Processing

- The U.S has maintained and updated the Monte Carlo Code MCNP to be consistent with the new format of FENDL/ENDF/B-VI data. *The MCNP is the reference design code for ITER and DEMO.*

- The U.S. is contributing to the development of the Fusion Evaluated Nuclear Data Library (FENDL) under the auspicious of IAEA. FENDL is the reference nuclear database for ITER.

- The U.S. is carrying out a major effort in benchmarking the newly developed FENDL data libraries by comparing calculated nuclear parameters to the experimental data of existing fusion benchmark experiments.

- The U.S. has developed and implemented techniques that treat pulsed operation mode in radioactivity inventory codes (e.g. RACC, DKR).

- Updating, maintaining, and adding additional transmutation reactions to REAC*3 and to FENDL activation libraries is a continuing effort by the U.S.

# Biological Dose Calculation

- An Important development effort is underway in the U.S. to provide reasonable methods and tools for calculating the biological dose outside the reactor building (as well as estimates of uncertainties). This is a very difficult problem that needs to be solved.

# INTEGRAL NEUTRONICS EXPERIMENTS AND ANALYSES FOR TRITIUM BREEDING, NUCLEAR HEATING, AND INDUCED RADIOACTIVITY

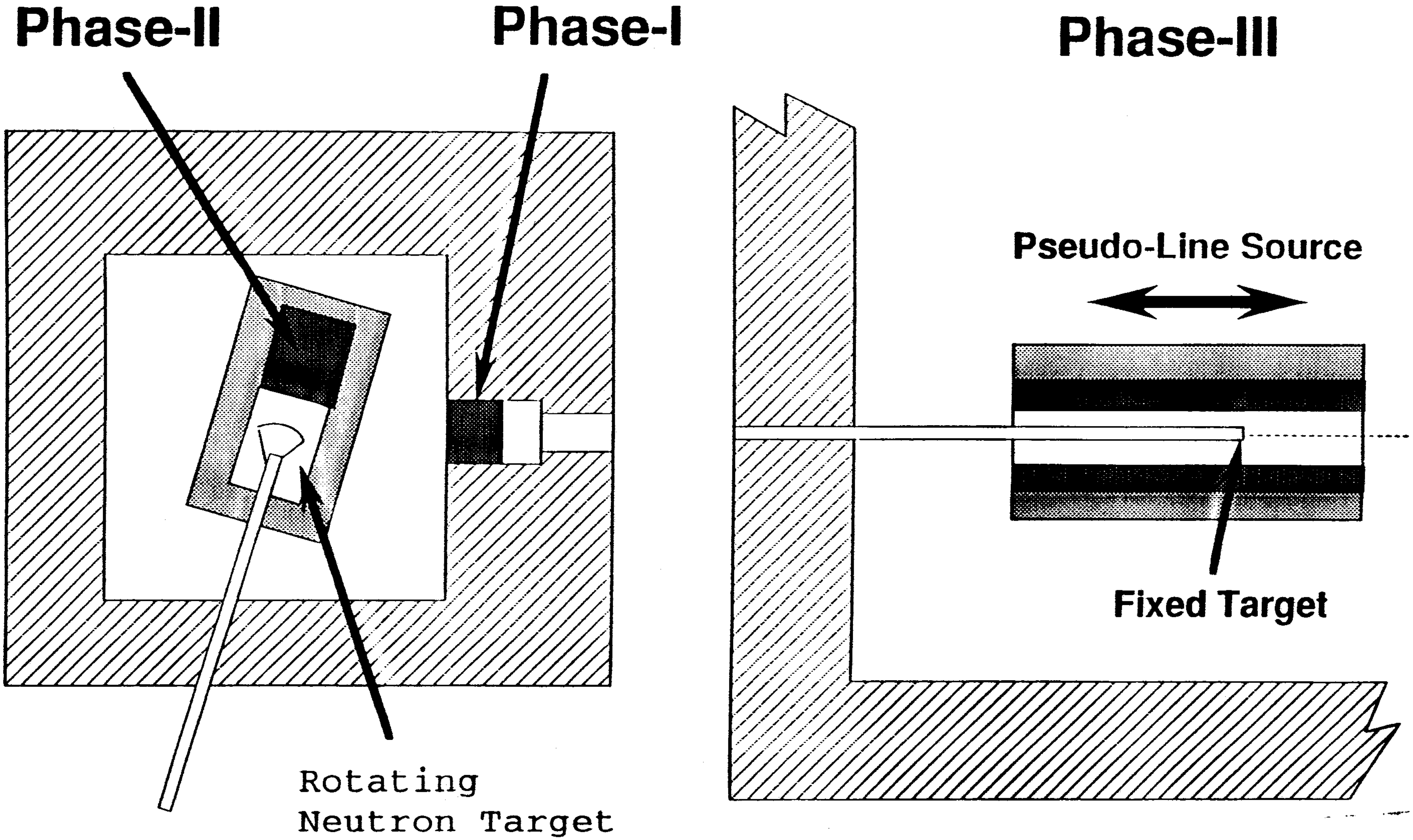
M. A. Abdou, H. Maekawa, Y. Oyama, M. Youssef, Y. Ikeda, A. Kumar, C. Konno, F. Maekawa, K. Kosako, T. Nakamura, E. Bennett

## ABSTRACT

A large number of integral experiments for fusion blanket neutronics were performed using D-T neutrons at the FNS facility as part of a 10 year collaborative program between JAERI and USA. A series of experiments were conducted using blanket assemblies that contained $Li_2O$, Be, steel, and water-coolant-channels with a "point" neutron source in a closed geometry that simulated well the neutron spectra in fusion systems. Another series of experiments were conducted using a novel approach in which the "point" source simulated a pseudo-line source inside a movable annular blanket test assembly, thus, providing a better simulation of the angular flux distribution for the 14-MeV neutrons. A number of measurement techniques were developed for tritium production, induced radioactivity and nuclear heating. Transport calculations were performed using 3-D Monte Carlo and 2-D Discrete-Ordinate Codes and the latest nuclear data libraries in Japan and USA. Significant differences among measurement techniques and calculation methods are found. To assure a 90% confidence level for tritium breeding calculations not to exceed measurements, designers should use a safety factor > 1.1-1.2, depending on the calculation method. Such a safety factor may not be affordable with most candidate blanket designs. Therefore, demonstration of tritium self-sufficiency is recommended as a high priority for testing in near-term fusion facilities such as ITER. The radioactivity measurements were performed for >20 materials with the focus on $\gamma$ emitters with a half-life <5 years. The ratio of the calculated (C) to experimental (E) value ranges between 0.5 and 1.5, but it deviates greatly from one for some materials with some cases exceeding 5 and others falling below 0.1. Most discrepancies are attributed directly to deficiencies in the activation libraries, particularly errors in cross sections for certain reactions. A microcalorimetric technique was vastly improved and it allowed measurements of nuclear heating with a temperature rise as low as 1 $\mu^{o}$K. The C/E ratio for nuclear heating deviates from one by as much as 70% for some materials but by only a few percent for other materials.

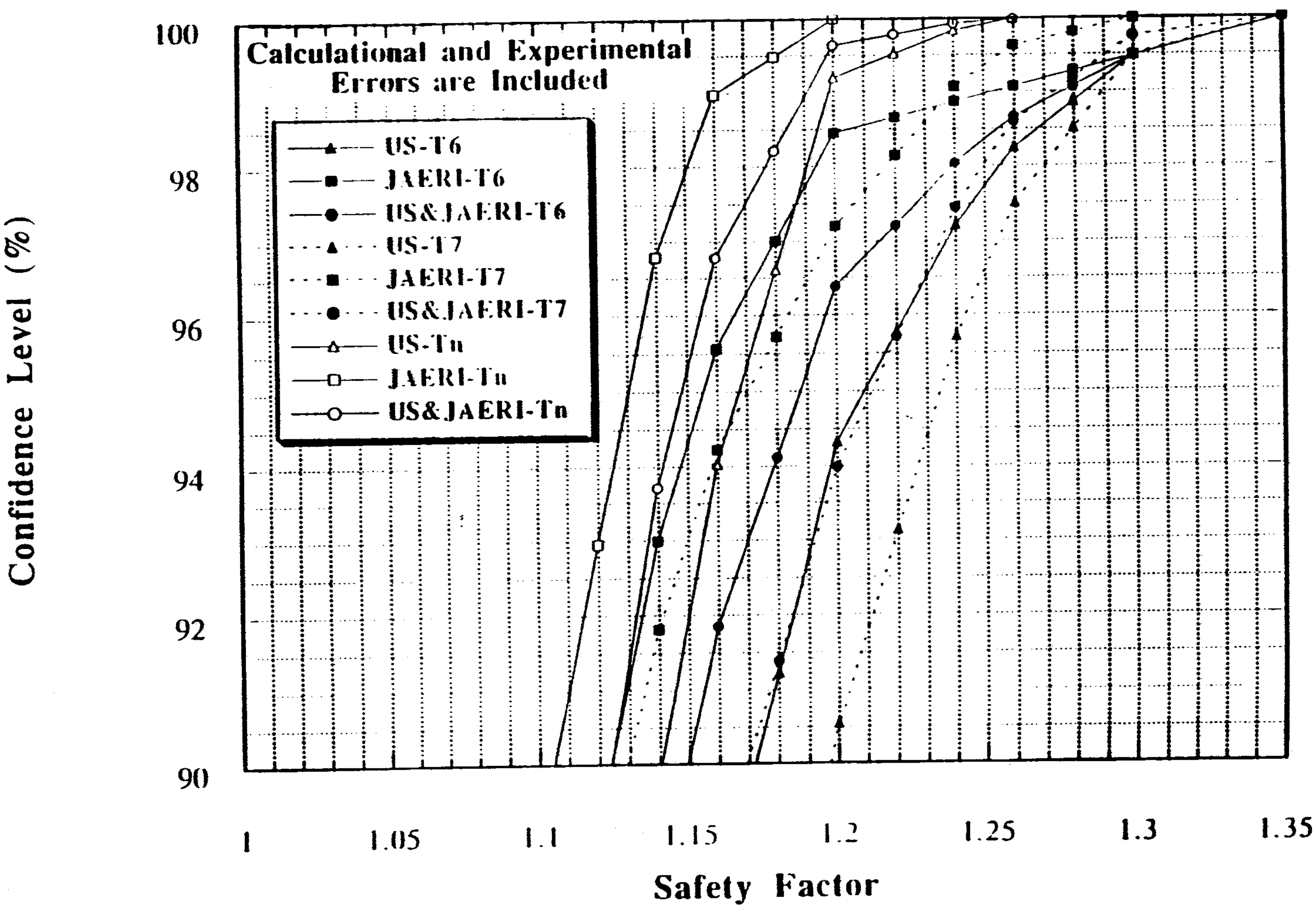
To appear in Special Issue of Fusion Technology, volume 28 August 1995

# Concept of Experimental Arrangement



# Design Safety Factor Versus Confidence Level for Tritium Production Rate

- (Confidence Level for Calculations not to Exceed Measurements)

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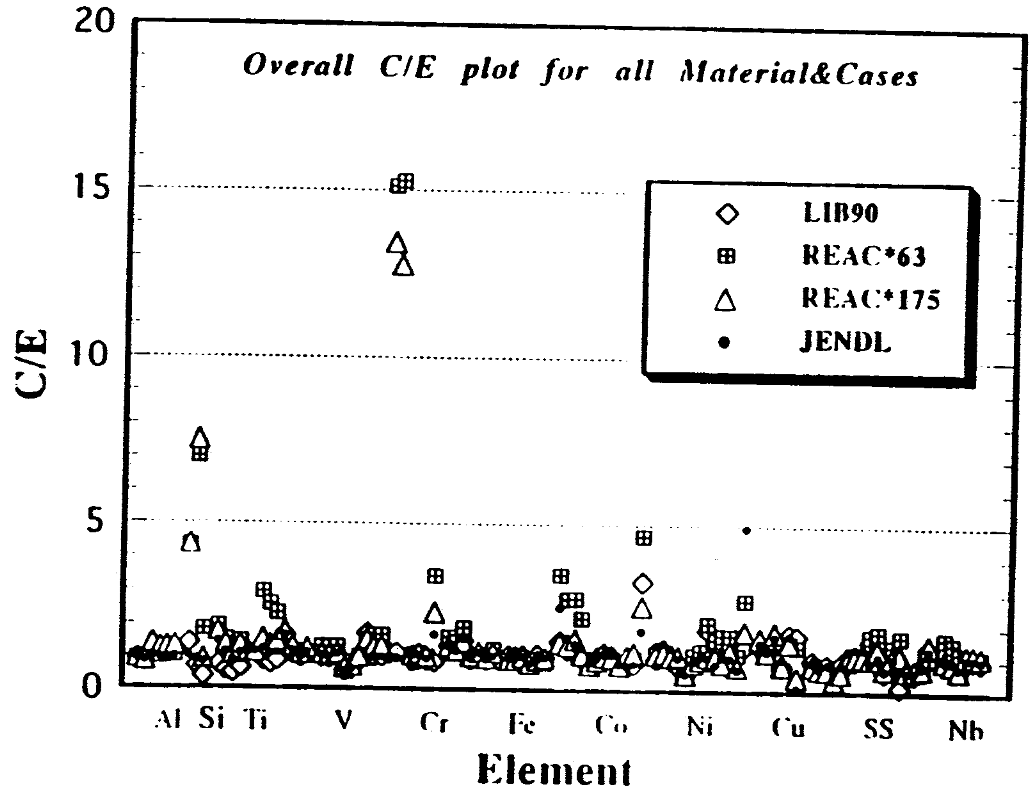
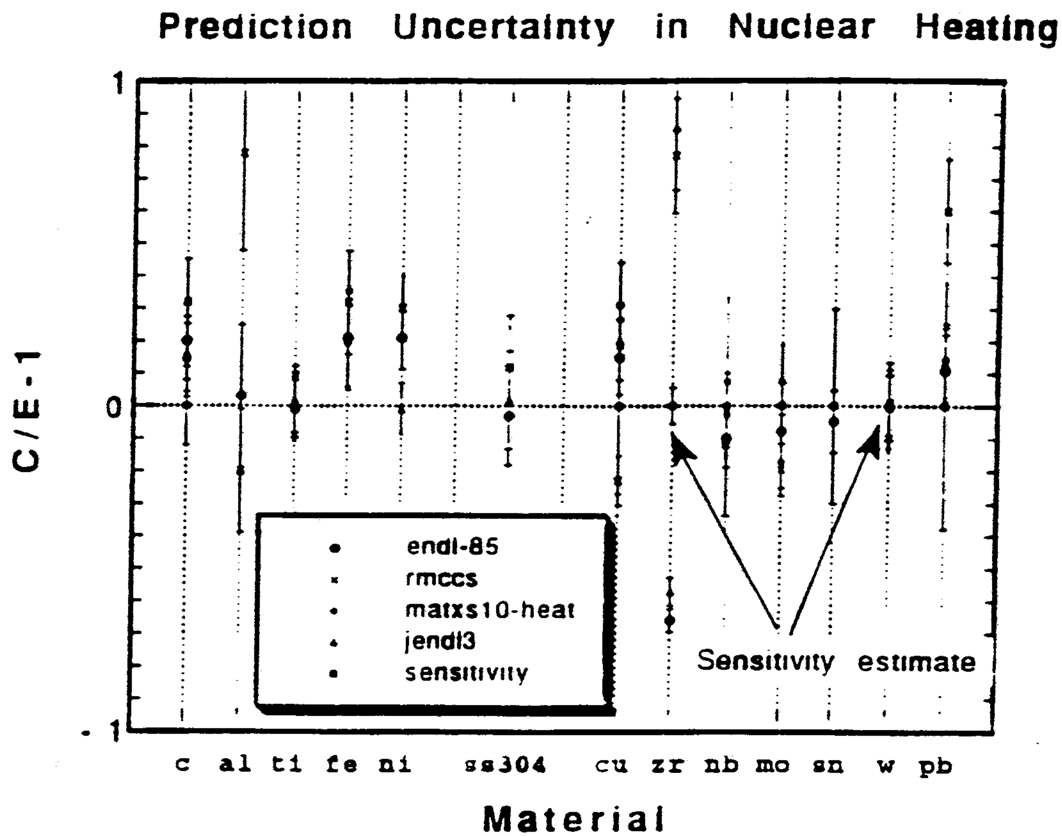
# Calculated-to-Experimental Values of Induced Radioactivity for Several Elements



- **Uncertainty in nuclear heating is unacceptably high**
- **20 to 30% uncertainty in nuclear heating will result in comparable uncertainty in operating temperature**



# Tritium Fuel Cycle

## R&D Activities

1) Dynamic Modelling of the entire fuel cycle
   UCLA, ANL

2) Tritium Extraction from Blanket
   ANL, LANL, UCLA

3) Tritium Plant Component Experiments
   TSTA: LANL

## Tritium Fuel Cycle Objectives

- Achieve FUEL SELF SUFFICIENCY
  i.e., ensure that the fuel cycle can be closed

- Safe and Reliable Operation
  » Minimization of Tritium Inventory
    - Economy and Safety Costs
  » Minimization of Tritiated Effluent Waste

- Optimization of Fuel Cycle Flowsheet Configuration

- Efficient Impurity Separation

- Efficient Hydrogen Isotope Separation

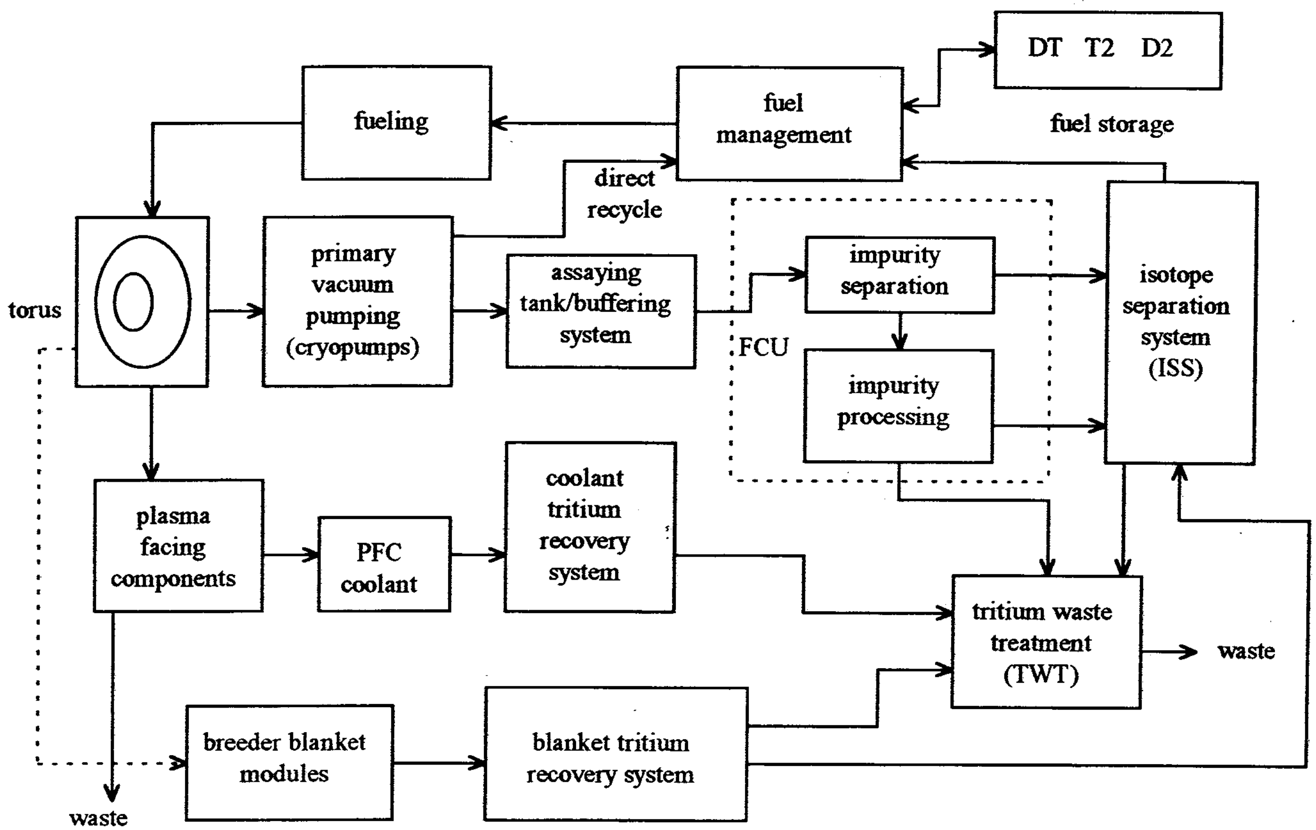
- Reliable Fuel Source Operation

- Efficient Blanket Breeder Tritium Processing

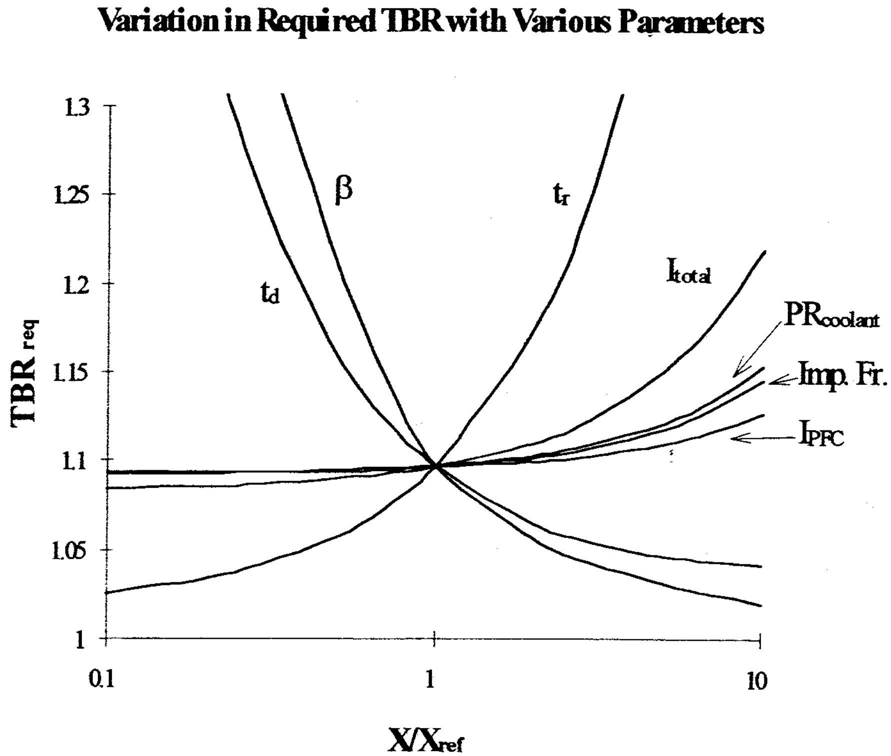
# TRUFFLES (TRitiUm Fusion Fuel cycLE dynamic Simulation)

- TRUFFLES development is near completion at UCLA
- TRUFFLES can help balance different design objectives; optimize the fuel cycle design
- Time-dependent tritium inventories and flow rates are accurately and efficiently calculated using TRUFFLES
- TRUFFLES is easy-to-use
- Modules are constantly being added and improved
- Tritium accountancy tool for Future Fusion Reactors
- Examination of <u>DT Fuel Self-Sufficiency Issue</u> for any type of Fusion Reactor [calculate <u>Required Tritium Breeding Ratio</u> as a function of reactor parameters ($\beta$, $t_d$, $\tau$, $\varepsilon$, etc.)

# TRUFFLES Fuel Cycle Flowsheet

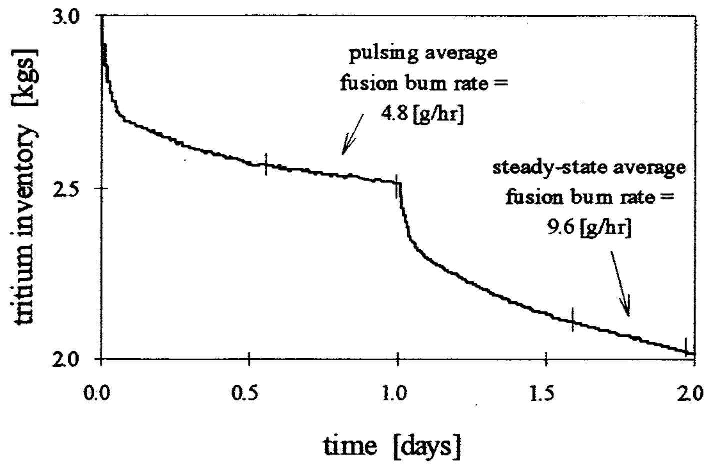


# Variation of Required TBR




- Parameter Reference Values:
  - » Doubling Time = $t_d$ = 5 [years]
  - » Fuel Fractional Burnup = $\beta$ = 2 [%]
  - » PFC Inventory = $I_{PFC}$ = 1 [kg]
  - » Total Tritium Inventory = $I_{total}$ = 4 [kg]
  - » Avg. Torus Exhaust Impurity Content = Imp. Fr. = 5 [%]
  - » Days of Reserve = $t_r$ = 2 [days]
  - » Permeation Rate into Coolant = $PR_{coolant}$ = 15 [g/day]

# Storage Dynamic Tritium Inventory




- The storage tritium inventory will depend upon the upstream residence time and the fusion fuel burnup (consumption)
- Characteristics:
  - » Overall upstream residence time of gas flows is short (~2 hours)
  - » Fusion fuel burnup, rate of tritium accumulation inside plasma facing components, and blanket breeder tritium generation will determine the long-term behavior