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FUSION ENERGY DIVISION

TOKAMAK BLANKET DESIGN STUDY:
FY 78 SUMMARY REPORT

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ACKNOWLEDGMENT

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ABSTRACT

A tokamak blanket cylindrical module concept was designed, developed, and analyzed after review of several existing generic concepts. The design is based on use of state-of-the-art structural materials (20% cold worked type 316 stainless steel), lithium as the breeding material, and pressurized helium as the coolant. The module design consists of nested concentric cylinders and features direct wall cooling by flowing helium between the outer (first wall) cylinder and the inner lithium containing cylinder. Each cylinder is capable of withstanding full coolant pressure for enhanced reliability. Results show that stainless steel is a viable material for a first wall subjected to 4 MW/m^2 neutron and 1 MW/m^2 particle heat flux. A lifetime analysis showed that the first wall design meets the goal of operating at 20 minute cycles with 95% duty for 10^5 cycles. The design is attractive for further development, and additional work and supporting experiments are identified to reduce analytical uncertainties and enhance the design reliability.

FOREWORD

This document summarizes work completed in FY 78 as part of a continuing long-range development program aimed at providing a blanket design concept for near-term application to tokamak fusion power reactors. Before proceeding into the details of this report, we provide some background for continuity with the previous work where objectives and guidelines were established. Additional background can be found in the Program Plan[1] for this year's work. We have also included in this foreword a statement of the results and conclusions in the context of the long-range program.

DEVELOPMENT OF STUDY OBJECTIVES AND GUIDELINES

Previous fusion reactor systems studies at ORNL have promoted the concept of a compact, high power density tokamak fusion reactor.[2] These studies have endeavored to establish a scientific, engineering, and economic basis for such a reactor. Specifically, we have pursued an approach that not only addresses the technical aspects of fusion technology, but also requires that we investigate the economic implications of adapting fusion for utilities application. In order for fusion to become a serious candidate technology for utilities application, it must, from the outset, be shown to be economically competitive with other advanced energy systems.

The findings of our previous systems studies have allowed us to pursue a rather specific approach to blanket design. The fundamental philosophy embodied in this approach required a critical evaluation of the functional merits of blanket designs in the context of their potential cost and the time necessary to develop them. For example, if viable blanket designs can be derived from existing technology, the substantial costs required for developing new technologies will be omitted. In addition, the lead time necessary to develop these technologies will also be avoided. However, to accomplish this in the design space provided by existing technology might require compromises in the performance of the blanket system that are less than optimal. On the other hand, the motivation for pursuing designs that utilize other than existing technology is to recognize some performance, reliability, or other design advantages that cannot be achieved with existing technology. One could argue that the costs of developing new technologies can be offset by decreased operating costs.

In this study, we have taken the design approach emphasizing the use of as much existing technology as is possible. For this reason, the study has been limited to consider generic blanket designs incorporating an austenitic stainless steel structure, lithium (liquid) moderator, and a gaseous coolant (helium). We argue that, at this time, this selection best represents our requirements of utilizing existing technology and satisfying functional engineering requirements. This selection reflects the conclusions of our previous parametric studies that have evaluated the relative merits of different combinations of structural materials, moderators, and coolants. It is not our intention to preclude other possible designs or material selections. Our objective was simply to provide a means by which we can focus the process of conceptual engineering one step further in comprehensiveness towards developing a feasible preliminary blanket design. Once this has been accomplished, then it seems appropriate that other competing designs could be meaningfully compared.

The methodology used in pursuing this current study was to involve an industrial subcontractor with special expertise in power plant technology. Its efforts were to be directed towards important and perhaps underaddressed design questions such as: tolerance for failure, structural response to thermal and magnetic transients, fabricability, maintainability, and lifetime. Westinghouse Electric Corporation was selected on the basis of its background in nuclear technology development and experience in fusion system studies.

Our original intention was to focus our design effort by selecting a promising candidate from generic designs that had been conceptualized previously. It was thought that by upgrading an existing design we could arrive at a design that was defensible not only with respect to function, but also with due regard to its reliability potential.

A critical review of previous designs indicated that no individual concept could stand up to the most fundamental considerations involving reliability and tolerance for failure. These considerations of reliability and tolerance for failure have precipitated what we consider to be the minimum design requirements for this generic type of blanket.

(1) It is essential that all critical blanket structure be actively cooled. This requirement arises because of the particular character of the heat deposition throughout all portions of the blanket and the uncertainty in surface contact between the stagnant lithium and the structural material.

(2) It is essential because of the potential for a leak in the high pressure coolant circuit that all enclosed regions of the lithium container be capable of sustaining the full coolant pressure without rupture or other catastrophic failure.

RESULTS AND CONCLUSIONS

It seemed apparent in the initial design review that all candidate designs satisfied the functional requirements for which they were designed. That is to say, the thermal, structural, neutronic and other functional requirements were, for the most part, satisfied. This situation was encouraging in that our efforts could be focused on developing a design that emphasized reliability, knowing in advance that conceptually the selection of structural material, moderator, and coolant was well founded with respect to function.

The new design resulting from this study, supported by detailed calculations, has again verified that conceptually there are no reasons for us to reconsider our selection of structural materials, moderator, and coolant. This study has shown:

(1) Positive margins of safety resulted from a detailed analysis of the blanket's structural performance. These calculations have taken into account cyclic thermal and magnetic loads, radiation damage, and creep. The predicted lifetime of the blanket modules is greater than 10^5 cycles or approximately four years of normal operation.

(2) Thermal-hydraulic performance is predicted to allow the achievement of approximately 31% gross thermodynamic efficiency while satisfying all structural temperature requirements.

(3) Although not yet established in detail, this design appears amenable to the development of plausible fabrication, maintenance, and tritium extraction schemes.

It is important to point out that in a conceptual design study of this type the results obtained are only as relevant as the assumptions used in developing the details of the study. We have, in this study, endeavored to use conservative judgment in areas where there is insufficient design data to precisely define problem areas and their design solution. Whether or not these judgments are indeed conservative will probably be the subject of some controversy. It is for this reason that we have made an effort to clearly delineate the uncertainty in specific design areas and in this way encourage constructive criticism. It is also important to indicate that the bulk of our design efforts have been directed toward the evolution of a basic module design with a lesser effort directed toward systems integration and optimization. Our efforts for FY 79 will involve considerably more attention to these important design considerations.

REFERENCES

1. P. B. Mohr (Oak Ridge National Laboratory), "Technical Program Plan for an Engineered Blanket Design," private communication with Charles Head, Department of Energy, Washington, D.C. (January 16, 1978).

2. D. Steiner et al., ORNL Fusion Power Demonstration Study: Interim Report, ORNL/TM-5813, Oak Ridge, Tennessee (March 1977).

1.0 SUMMARY

This report summarizes the results of a study, completed in FY 78, which is part of a continuing Tokamak Blanket Design Program. The objective of the FY 78 program was to select a reference blanket design concept, develop the design supported by adequate analysis, and assess the performance of the design. As a result, recommendations for future design tasks and research and development programs to support future continuing work were developed.

The reference concept selection was limited in scope to consideration of blankets which incorporated specific characteristics. The blanket structure was specified as stainless steel, while the tritium breeding material was specified as liquid lithium. Cooling was to be achieved by pressurized helium. For the purposes of comparison, existing blanket designs were categorized into three generic concept categories. These three generic concept categories are discussed in Section 3.0.

A set of selection criteria was generated and mutually agreed to by ORNL and Westinghouse. As a result of the review of existing concepts against the selection criteria, additional key design requirements (which are significant in guiding the design of a blanket for power reactor application) were identified. These requirements are:

- From a reliability standpoint, the blanket module should be capable of withstanding full coolant pressure since it is judged that a coolant leak cannot be precluded.
- To assure reliability, it is necessary that the lithium containment structure (module body) incorporate integral cooling circuits to preclude burnout related to bubble formation or other thermal unbonding.

A review of existing concepts indicated that none could simultaneously satisfy these criteria. In the process of evaluating designs against these key requirements, a cylindrical module concept was evolved and was recommended as the concept which was developed in this study. This concept consists of an outer cylinder (with a spherical nose first wall) surrounding an inner lithium containing cylinder with helium flowing between these concentric cylinders to achieve adequate cooling of both the lithium and the outer first wall. A schematic illustration of the module is shown in Figure 1.0-1.

The selected concept was developed considering in greater detail fluid (pressure) and thermal loads, considerations for assembly/disassembly and remote maintenance, and removal of the lithium for recovery of generated tritium. In addition, the removal of generated helium was addressed by providing a feature in the design for venting of the helium. In the process of developing the design, manufacturing feasibility was considered. The cylindrical module proved to be extremely attractive based on its simple, structurally efficient shape and relative adaptability to being mass-produced. Modules can be packaged as replaceable subassemblies with potential for reasonable assembly and maintenance.

A 10.2 cm (4 inch) diameter, 75 cm (29.53 inch) long module was selected as a reference case for which detailed supporting thermal and structural analysis and lifetime assessment were made based on the following parameters for normal operation:

- Helium Coolant Pressure — 5.5 MPa (54.4 atm)
- Helium Inlet Temperature — 200° C
- Maximum First Wall Temperature — 450° C
- Neutron Wall Loading — 4 MW/m^2
  (First Wall Heat Flux — 1 MW/m^2)
- Pumping Power — $\leq$ 2-2.5% of Gross Thermal Output.

The detailed thermal analysis of the module provided temperature distributions which permitted a detailed structural analysis of the outer cylinder of the module, particularly the first wall, which is the most critical element because

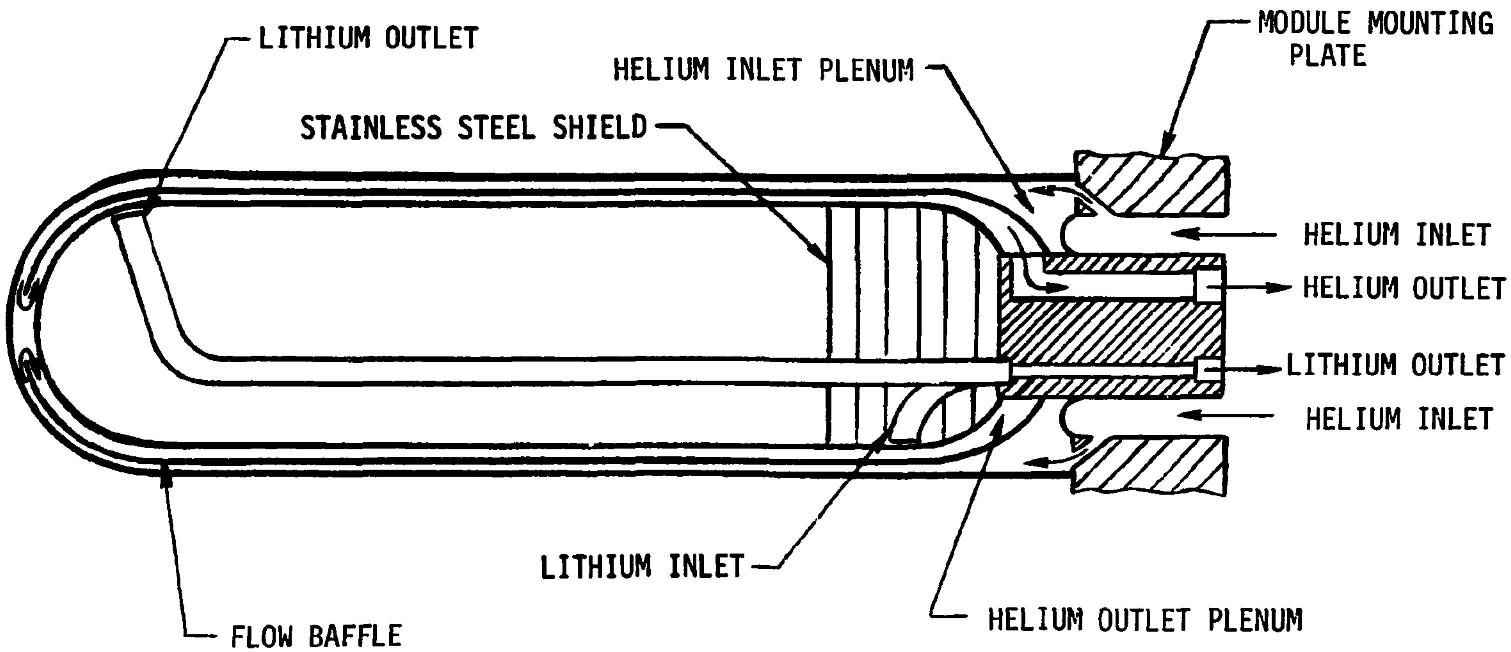

Figure 1.0-1. Schematic of Cylindrical Module Concept

of the high neutron loading and heat flux. The stresses calculated by the ANSYS finite element computer code were used to calculate the margin of safety for lifetime considerations based on crack growth and brittle fracture. In addition, a preliminary calculation of the potential effect of swelling that might lead to outer module growth, which could influence the coolant flow gap, was considered. When assessment of the design was performed considering normal operating conditions, a positive margin of safety was obtained for each of the potential failure considerations with the largest margin of safety for crack growth, then for brittle fracture, while the potential deformation had the smallest but still positive margin.

Based on the results of the design study and supporting analyses presented in Section 4, the cylindrical blanket concept is a viable approach and should continue to be developed. The following conclusions form the basis for recommending that the concept be further developed.

- A stainless steel blanket assembly with the cylindrical modules represents a design concept which can meet the temperature, lifetime (10^5 cycles of 20 minutes duration at 95% duty), and thermal requirements of the guidelines for the study and still provide an adequate breeding ratio.
- The design concept is simple from a configuration standpoint and attractive for both analytical and experimental evaluation.
- The design is structurally efficient, is amenable to mass production, and can be readily fabricated.
- The design is capable of being constructed as replaceable assemblies consistent with the philosophy for reasonable assembly and maintenance.

Some of the disadvantages in this concept are:

- Designing the modules to withstand full coolant pressure results in many small modules which require piping and connections for the coolant and lithium circulating system.
- The voids between the stacked cylinders require longer modules (resulting in a thicker blanket assembly) to obtain the effective lithium breeding volume desired.

- The large number of modules may impact the system operating reliability and require more quality control inspection, checkout, and time to assure that reliable components and assemblies are fabricated.

A design assessment of the module for other than normal operating conditions was performed. These included: off-design conditions, changes in cycle time, and incorporation of a divertor. In addition, the effects of loss of coolant and hypothetical plasma disruption were addressed. The results of this assessment with predicted margins of safety based on lifetime assessment are presented in Section 5.0 and summarized in Table 5.1-1. Some of the key observations resulting from the assessment are as follows:

- The structural lifetime margins of safety for a postulated 10% reduction in helium coolant flow are essentially unchanged and a slight improvement in thermal performance is achieved compared to normal design operating condition.

- At one-half power (2 MW/m^2 neutron flux) the margins of safety are substantially increased and the pumping power reduced from 2.2% to less than 1%.

- If the duration of the pulse is decreased by a factor of 10, the margin of safety relative to crack growth is decreased by a factor of 10 (but still acceptable) for the same total operating time while the thermal performance is slightly reduced. Conversely, if the duration of the pulse is increased by a factor of 10, the margin of safety relative to crack growth is increased by a factor of 10 and the thermal performance slightly improved.

- For the case of a 100% efficient divertor the margins of safety are virtually infinite from crack growth and excessive growth considerations and increase by a factor of $\sim$ 3 relative to brittle fracture for a normal duty cycle (20 minutes, 95% duty, for 10^5 cycles). The pumping power is reduced substantially (from 2.2 to 1.25%).

- For a hypothetical plasma disruption based on a 0.010 s plasma thermal pulse, incipient melting of the first wall can occur unless the heat flux is limited to approximately 20 MW/m^2.

- A postulated loss of coolant could be tolerated only if a rapid shutdown ($\sim$ 400 ms) is initiated after loss of flow.

The cylindrical blanket concept design should be continued based on the encouraging thermal performance and structural lifetime analysis results. Since the effort to date has focused primarily on developing a reliable module with attractive thermal and structural lifetime performance, less design effort was devoted to incorporating the module into an overall blanket system. This should be done in a follow-up program. Consistent with this philosophy, recommendations for future work are proposed in Section 6.0 to address some of the areas where improvement or verification of analytical assumptions might be achieved. These key recommendations are as follows:

- Consistent with structural and thermal performance, increase the module size and decrease system complexity to enhance breeding and reliability.
- Develop the blanket system design in sufficient detail to permit an assessment of the blanket system reliability and perform a reliability assessment relative to system performance capability for operation in a reactor environment.
- Perform detailed neutronics analysis to verify tritium breeding capability.
- Calculate the reflectivity of higher modes of cyclotron radiation to determine if the small module arrangement has any potentially adverse effect on plasma temperature and heating of the sides of the modules.
- Evaluate the design to determine compatibility with high vacuum techniques.
- Implement test programs to generate data to confirm assumptions used in the analytical assessments of thermal performance and structural lifetime assessment of the blanket.

2.0 INTRODUCTION

This report presents the results of a joint Westinghouse/ORNL program. The purpose of this program is to produce a reference design concept for a tokamak blanket system that will operate under reactor conditions and to assess the performance and lifetime aspects of the design. The resulting concept is based on state-of-the-art materials and manufacturing technology and was limited at the outset to use stainless steel structures, lithium as the tritium breeding material and helium as the coolant. The study is intended to advance the state of the art of blanket concepts a step closer to a design for power reactor application. The study was performed in three phases which consisted of selecting of a design concept, developing the design concept justified by supporting detailed analysis, and assessing the developed design against the design requirements.

The first phase of the program, selecting a design concept, consisted of reviewing existing generic candidate blanket concepts, assessing them against defined criteria, evolving key design requirements, and establishing and recommending a reference blanket concept. The resulting design selection is documented and reported in Section 3.0 of this report.

In the second phase, the selected concept was developed, supported by detailed thermal-hydraulic analysis and structural and lifetime analysis considering the requirements and constraints specified in the concept selection phase. The design considered manufacturing feasibility and considerations for assembly/disassembly and maintenance. The results of this effort are documented in Section 4.0.

In the final phase, an engineering assessment of the design was performed. The considerations included: assessment under nominal and off-design conditions, consideration of loss of coolant and part power operation, assessment of design with increased and decreased cycle time (pulse length), consideration of the impact of incorporation of a divertor, and assessment of routine service and maintenance requirements.

The assessment of the design against normal operating conditions is presented in Section 4.0. Off-design conditions, incorporation of a divertor, and service and maintenance requirement assessments are documented in Section 5.0.

Finally, conclusions resulting from the study are presented and recommendations are made for future effort in the areas of design and development, including test experiments to support future continuing effort in the blanket program. These conclusions and recommendations are contained in Section 6.0.

3.0 CONCEPT EVALUATION AND SELECTION

An engineering review and assessment of existing design concepts was performed as the first phase of this study. Design guidelines and evaluation criteria for generic concept selection were established and mutually agreed to by Westinghouse and ORNL as an aid to focus the study. Additionally, during the assessment of concepts reviewed, key guidelines were identified which led to key requirements upon which to base a concept selection.

3.1 KEY DESIGN GUIDELINES

Based upon earlier blanket design work,[1] a number of key guidelines (Table 3.1-1) were established to aid in focusing the current study. These included:

- Selection of austenitic stainless steel for the primary structure.
- Selection of helium as the coolant.
- Selection of natural liquid lithium as a tritium breeding medium.

The blanket structural material must operate reliably, must contain the coolant and breeding medium, must maintain the required purity in the plasma chamber, and must be replaceable in the event of a failure. The structural material must operate in a cyclic loading environment and must operate at heat transfer temperatures adequate for attractive power conversion. Twenty percent cold worked (CW) type 316 austenitic stainless steel (316 SS) was selected as the primary structural material. This selection was based on the fact that the radiation response characteristics of this alloy are fairly well known at the temperatures of interest in this program. In addition, 20% CW-316-SS has been chosen as a reference material for the fusion energy Alloy Development for Irradiation Performance (ADIP) program.

TABLE 3.1-1

KEY DESIGN GUIDELINES

Breeding Medium	Lithium
Structural Material	Type 316 Austenitic Stainless Steel
Coolant	Pressurized Helium
Structural Concept	Modular
Vacuum Enclosure	External to Blanket
Structural Material Temperature Limits	∿ 400° C, First Wall (Radiation Damage Zone)
	∿ 500° C, Low Radiation Zone
	∿ 550° C Maximum — Nonstructural
Neutron Wall Loading	2-4 MW/m^2
First Wall Particle Heat Flux	0.5-1.0 MW/m^2 (without divertor)
Coolant Outlet Temperature	High as practical, consistent with meeting material structural limits
Pumping Power, $\left(\frac{\text{Pump Work}}{\text{Thermal Output}}\right)$	< 2%
Tritium Breeding Ratio	1.2
Duty Cycle	20 minute Cycle, 95% Duty
Operating Mode	Pulsed to 10^5 Cycles

Recognizing the materials property changes that occur with irradiation[2] at elevated temperature, guideline limits were established for various sections of the blanket. The guideline for the maximum temperature of the first wall was set at ∿ 400° C, the maximum temperature for lower stressed regions was set at 500° C, and the maximum temperature for any point in the structure was set at 550° C.

Helium was tentatively selected as the coolant because it represents an existing technology, has no adverse effect on breeding, is electrically nonconducting, presents no MHD problems, has low neutron absorption and is compatible with the other blanket materials. It is recognized that high coolant pressures are required and careful design is necessary to limit the pumping power demands.

Natural lithium metal was selected as the breeding medium based on its breeding capabilities and heat transfer properties.

A secondary vacuum enclosure[1,3-5] was identified as part of the system to eliminate the requirement for making a high vacuum tight joint for assembly and disassembly of the blanket. The guideline for lifetime considerations was a 20 minute cycle, 95% duty (19 minutes on, 1 minute off) and 10^5 pulse cycles.

3.2 EVALUATION CRITERIA

In the process of selecting the design concept developed in the study, it was necessary to establish the criteria to be used to assess candidate concepts. Accordingly, a set of five important criteria was developed to provide a consistent basis for evaluation. These are presented in Table 3.2-1 in order of importance.

TABLE 3.2-1

BLANKET CONCEPT SELECTION DESIGN CRITERIA

Reliability
Failure Tolerance
Maintainability
Fabricability
Incremental Performance

In order to meet the objectives of a reasonable plant availability for a demonstration reactor, it is imperative that the components perform with a minimum of failures. This can only be accomplished by achieving a high degree of reliability in the performance of components. For this reason, reliability was identified as the most important criterion in the selection process. Next to reliability, tolerance for failure and maintainability were considered about equally important. Fabricability, which included cost, was ranked next, but it was judged that this should not be as highly weighted as the previous three items for a number of reasons. Each concept, although not representing a complete design, was judged capable of being fabricated. Cost, at this time, was not judged to be a prime consideration since it was judged more important that a concept be developed with a good chance of meeting reliability and performance. Once these are demonstrated, it was judged that the design could be engineered to be more cost-effective. It was considered that incremental performance, the last of the criteria, should not be as heavily weighted as the others. It was assumed that each candidate must meet the basic performance requirements within the design guidelines; otherwise, it would be rejected. Once the basic performance, which is a major consideration, is met, then it is logical to attach some (but a lesser) weight to the criterion of incremental performance, which considers how well the concept performs beyond the specified minimum guideline performance level.

3.3 GENERIC CONCEPTS REVIEWED AND ASSESSMENT OF CANDIDATE DESIGNS AGAINST CRITERIA

The design concepts which were reviewed were characterized as being of three generic concepts: unpressurized, pressurized, or a combination (hybrid) of the two. The unpressurized concept is defined as one in which the helium cooling circuit is pressurized, but the lithium absorber module is incapable of withstanding any substantial pressure and is unpressurized or at a low pressure. For the pressurized concept, the entire blanket module (including the lithium or absorber) is designed to withstand the full coolant pressure. In the combined (hybrid) concept, the coolant conduits are constructed to form module walls in the form of tubes, double walls, tube and wall, or other similar combinations. The lithium volume is not necessarily pressurized. Following is a list of specific concepts reviewed within the three generic categories. These concepts are not discussed; however, the appropriate references from which the concept designs and analyses were extracted are identified.

UNPRESSURIZED CONCEPTS

- The ORNL EPR module design[6]
- The ORNL DEMO module design[1]
- The ORNL EBT module design[7,8]
- The Culham module design by Mitchell and Booth[9]

PRESSURIZED CONCEPTS

- The General Atomic DEMO blanket module[10]
- The Westinghouse actinide burner blanket concept[11]
- The Culham module design by Mitchell and Booth (circulating lithium cooled)[9]

HYBRID CONCEPT

- ORNL cassette blanket concept[3,12]

Since many of the concepts reviewed had significant differences in design parameters and characteristics, Table 3.3-1 was generated and is included to provide a convenient comparison between design concepts. The cylindrical concept, which evolved during this task and is discussed in detail in Section 3.5, is included in the table for information.

As a consequence of performing the concept review, Table 3.3-2 was prepared. This table is a listing of advantages and disadvantages associated with each of the concepts and a judgmental evaluation of each of the concepts with respect to each item listed.

In the process of performing the review of existing design concepts and in generating the criteria identified in Table 3.1-1, a few key problem areas and design requirements were identified. These requirements are discussed in the following section.

TABLE 3.3-1

COMPARISON OF BLANKET MODULE PARAMETERS AND CHARACTERISTICS

PARAMETER	UNITS	ORNL EPR[6]	ORNL DEMO[1]	ORNL EBT[7,8]	CULHAM (MITCHELL & BOOTH)[9]	GENERAL ATOMIC DPR[10]	WESTINGHOUSE ACTINIDE BURNER[11]	ORNL CASSETTE[1,3,12]	CYLINDRICAL MODULE
First Wall Particle Heat Flux	MW/m^2	(b)	0.5-1.0[g]	0.37	$\gtrsim$ 2/0.5[a]	0.45	0.43[b]	0.7-1.0	0.5-1.0
Neutron Wall Loading	MW/m^2	1.0	2-4	1.47	8/4[a]	1.85	1.15	3-4	2-4
Module Internal Pressure	Atm (Psi)	Est 1-2 Atm	Est. < 1 Atm	.07 (1.0)	U	50 (735)	70 (1030)	Est $\gtrsim$ 1	$\gtrsim$ 1
Coolant Pressure	Atm (Psi)	70 (1030)	Low Press.	($\geq$ 1000)	U	50 (735)	70 (1030)	20-80 Analyzed for 60	54.4 (800)
Wall Material	—	316 SS	SS	316 SS	Niobium	Inconel 718	316 SS[b]	316 SS	316 SS
Pumping Power $\frac{\text{Pump Work}}{\text{Thermal Output}}$	%	6.7	U	1.5-3	< 3/U[a]	3-5 Target	2.0	1-2	< 1.5
Coolant	—	He	HITEC	He	He/Li[a]	He	He	He or HITEC	He
Breeding Ratio	—	~ 1.15	> 1	1.29	1.3[a,f]	$\geq$ 1	N/A	U	Est >1.1
First Wall Temp.	° C	(b)	400° Max	400° Max	$\frac{505\text{-}650°}{466\text{-}538°}$[a]	$\gtrsim$ 600° Max.	450° Max[b]	400-500°[d]	450°
Wall Thickness	in (cm)	0.25 (0.64)	U[g]	0.125 (0.32)	0.18 (0.45)	0.197 (0.5)	(2.0)[b]	(0.175)Tube[c]	0.062(0.16)
Coolant Temperature	° C	200-370°	450-500°Max	66-481°	$\frac{355\text{-}650°}{350\text{-}650°}$[a]	275-585°	150-845°	$\gtrsim$80-480°	200-435°
Breeding Material	—	Li	Li	Li	Li	Li_7Pb_2, Li_4Si_4[h]	N/A	Li	Li
Tritium Recovery	—	U	U	Drain Module	$\frac{U^{(a)}}{\text{Circ. Li}}$	U[h]	N/A	Nb Window & Capillary	U

(a) For Culham Module

Number in Numerator for Helium Cooled Module
Number in Denominator for Circulating Li Cooled Module

(b) First Wall Not Integral with Module

(c) For 4 MW/m^2 Neutron Wall Loading and 1.5% Pumping Power

(d) First Wall Side and Li Side Respectively

(e) In One Experimental Module

(f) Utilizes graphite reflector

(g) Separate Tubular First Wall

(h) Uses Solid Li Alloy Rods

U Unknown or Unspecified

N/A Not Applicable

TABLE 3.3-2

SUMMARY OF ADVANTAGES AND DISADVANTAGES

ADVANTAGES/DISADVANTAGES	UNPRESSURIZED				PRESSURIZED		HYBRID
	EPR	ORNL DEMO	EBT	CULHAM (HELIUM GAS CIRCULATION)	GENERAL ATOMIC	(W) ACTINIDE BURNER	ORNL CASSETTE
ADVANTAGES - (LEGEND RATINGS ○ GOOD ⊗ FAIR ● BAD)							
Close packing of blanket modules	○	○	○	⊗	⊗	○	○
Large modules, less plumbing fewer units to work with	○	○	○	●	⊗	○	⊗
Double wall construction between lithium and plasma	NA	○	NA	NA	NA	NA	○
Light weight, stable configuration	NA	NA	NA	○	○	NA	○
Standardization of components	●	●	●	○	○	●	○
Machine made welds	⊗	⊗	⊗	○	NA	⊗	○
Independent expansion of lithium chamber and first wall	NA	⊗	NA	NA	NA	NA	○
Lithium/metal volume ratio	○	○	○	⊗	⊗	○	○
DISADVANTAGES - (LEGEND RATINGS ● HIGH ⊗ MED ○ LOW)							
Flat walls subject to thermal stress	●	●	●	NA	NA	●	⊗
Corner stresses	●	●	●	NA	NA	●	⊗
Variety of sizes and shapes required	●	●	●	⊗	NA	●	⊗
Cooling tube header welds with difficult geometry	NA	⊗	NA	⊗	NA	NA	●
Thermal stresses due to different wall temperature	●	○	●	NA	⊗	⊗	⊗
Hot spots created by helium bubbles generated within the lithium	●	○	●	●	NA	NA	○
Difficult tube bending arrangement	●	●	●	⊗	NA	NA	NA
Differential expansion between inner and outer walls	NA	⊗	NA	NA	NA	⊗	○
Interstitial inserts required	NA	NA	NA	⊗	●	NA	○
Difficult bellows fabrication	NA	NA	NA	NA	NA	NA	●
Welds in high radiation fields	●	●	●	⊗	NA	●	●
Neutron streaming	○	○	○	○	⊗	○	○
Corrosive effects of coolant	NA	●	NA	NA	NA	NA	NA
Difficulty of replacement	⊗	⊗	⊗	⊗	●	⊗	○

3.4 DEVELOPMENT OF KEY DESIGN REQUIREMENTS

The review and evaluation of existing concepts led to the identification of problem areas which may be significant. These problems include:

- The potential adverse impact on heat transfer resulting from accumulation of generated helium between the first wall and the lithium heat transfer medium.
- The critical problem of effectively removing high heat loads from the walls by heat transfer through lithium.
- Pressurization of the module in the event of a cooling circuit failure.
- The potential for significant adverse magnetic loads.

3.4.1 WALL COOLING CONSIDERATIONS

Designs which achieve wall cooling by transferring the heat from the walls through the lithium and then through coolant tubes to the helium coolant appear to become impractical as the particle heat flux level is increased. Most of the existing designs were based on particle heat fluxes of less than one-half of the 1.0 MW/m^2 guideline established for this study. The series of thermal impedances between the plasma and the helium coolant requires excessive pumping power or excessive coolant pressures if material temperatures are to be maintained within the specified maximum limits. Any further thickening of coolant tube walls to accommodate higher pressure increases the thermal impedance of the tube and penalizes tritium breeding by increasing the volume percent of the metal within the module. Larger diameter tubes to provide more flow without compromising pumping power also have the undesirable effect of moving the coolant away from the first wall, since the distance for heat flow from the first wall to the center of the coolant passage is increased when a larger diameter tube is used.

Some of the existing designs considered particle heat flux loadings in the range of those specified for this study and proposed cooling the first wall directly by the helium instead of attempting to transfer the heat to the coolant through the breeding material and coolant tube wall. The current review indicates that the higher particle heat flux level is a forcing function that leads to direct or integral cooling of the module walls.

As part of the tritium breeding process, helium gas can be generated in quantities sufficient to expect the helium gas to accumulate. In modules located near the bottom of the torus, this helium gas could collect between the first wall and the lithium and create a gas pocket that would present a high thermal impedance to heat removal and would seriously inhibit heat removal from the first wall if such heat had to be transferred through the lithium to a coolant tube immersed in the lithium. In the extreme, this could lead to wall burnout. However, by employing an integrally cooled first wall, wall burnout potential would be greatly reduced. From considerations of difficulty of achieving heat removal via longer thermal flow paths and the adverse effect of accumulated helium on heat transfer, an integrally cooled wall is considered to be a necessary requirement for a module design at the particle heat flux levels under consideration.

3.4.2 MODULE PRESSURIZATION CONSIDERATIONS

When existing concepts were reviewed from reliability considerations, it was apparent that a postulated leak from which lithium could escape and enter the plasma or reactor cell was unacceptable. Although duplex cooling tubes have been proposed[7] to minimize the possibility of a helium coolant leak into the module, design uncertainties based on the present state of the art in blanket design make it difficult to preclude the possibility of a coolant leak. Since the breeding material may not be capable of being vented overboard at a sufficiently rapid rate, any significant coolant leak would pressurize the lithium module. To prevent overpressurization and subsequent rupture of the module, it was concluded that at this stage in the design of blanket modules for power reactor application, it is prudent to require that a module be designed to sustain full coolant pressure. It is recognized that a less conservative approach may prove viable once the fusion program develops experience with blanket operation, but to assure reliability of blanket design at this point in the program, it was judged necessary that the blanket module be designed to carry the full coolant pressure.

3.4.3 MAGNETIC CONSIDERATIONS

An area of potential significance which surfaced during the concept review was that of magnetically induced forces and torques which may be transmitted to the lithium-containing module when the poloidal field is pulsed. When the poloidal field is pulsed, the resultant body forces depend on a number of interacting factors including the strength of the toroidal magnetic field, the time rate of change of the poloidal field, the viscosity of the lithium, the size of the blanket module, and the location and orientation of the module in the reactor. A simplified analysis, discussed further in Section 4.5, indicates that magnetically induced loads vary as the third or fourth power of the module characteristic dimension, where the characteristic dimension is defined as the radius or half width of a long module of circular or square cross section, respectively. This suggests that care must be exercised in selecting the blanket module size to assure that whatever forces are generated can be accommodated by the design.

Another factor to be considered is the effect of the induced circulating currents within each module on the pulsed poloidal field. The pulsing poloidal field is required to maintain the plasma in equilibrium; the induced circulating currents in the fluid produce fields which tend to oppose the pulsing field, and the larger the module, the more effectively the poloidal field is opposed. Therefore, from electromagnetic field design considerations, it is essential to assure that the design permits the pulsed field to penetrate the blanket module.

3.4.4 SUMMARY OF KEY DESIGN REQUIREMENTS

In summary, the review of existing designs with the foregoing considerations led to the following significant conclusions resulting from this evaluation.

- From a reliability standpoint, the lithium module should be capable of withstanding the full coolant pressure, since a pressure boundary leak cannot be precluded.

- Integrally cooled walls (cooled directly by the helium) are required to permit effective heat transport from the first wall at the heat flux levels being considered.
- Smaller modules are desired to reduce magnetically induced loads and to avoid suppressing the essential pulsing magnetic field.

These conclusions were factored into the requirements for blanket module design concept selection discussed in the following section.

3.5 EVOLUTION OF DESIGN CONCEPT

Incorporation of the key design requirements into the selection criteria led to the following observations relative to existing design concepts.

3.5.1 UNPRESSURIZED CONCEPTS

Although the unpressurized designs might be modified to incorporate integral wall cooling (which was a feature of one concept) and their basic size could be changed, if required, there appeared to be no reasonable design solution which would permit the module to sustain full coolant pressure. All of the unpressurized concepts reviewed featured rectangular modules. Such a configuration, with flat sides, is inherently inefficient from a pressurization standpoint. There appeared to be no way of accommodating a high coolant pressure on the module structure without resorting to unduly heavy walls.

3.5.2 PRESSURIZED CONCEPTS

The pressurized concepts examined, as currently defined, used liquid lithium coolant or helium coolant with solid compounds for breeding and were not readily adaptable to the ground rules. The pressure-carrying capability of a spherical or semielliptical first wall is an attractive feature of these designs. Although these candidate designs did not strictly satisfy the guideline requirements for a liquid lithium absorber with pressurized helium coolant, a pressurized concept is considered highly desirable, if integral wall cooling can be incorporated.

3.5.3 HYBRID CONCEPT

The attractive feature of the hybrid is that by definition, the coolant circuits form the external walls of the module. This feature, therefore, meets the design requirement of an integrally cooled wall to accommodate the high first wall heat flux and minimizes the effect of helium accumulation within the module, since the outer wall is cooled directly by the coolant and does not rely on heat transfer through the lithium. Existing concepts, however, employ rectangular modules with basically flat walls and do not have adequate capability for withstanding internal pressure within the module itself.

3.5.4 DEVELOPMENT OF AN ALTERNATE CYLINDRICAL CONCEPT

Because the existing concepts did not meet the design requirements developed or established during the first phase, an alternate concept was developed to address the areas of concern previously identified: specifically, pressurization of the module due to coolant circuit failure, effect of helium gas generation on heat transfer, and high heat removal requirement of the first wall. Additionally, the effect of magnetically induced loads was considered in developing the concept.

The candidate concept consists of cylindrical modules positioned approximately radially around the plasma with a spherical nose first wall facing the plasma. The modules are nested together, on a triangular pitch, as closely as is practical.

The module concept is shown schematically in Figure 3.5-1 and consists of a double wall stainless steel cylinder with spherical ends designed to carry the high pressure helium coolant. For the reference case, the outside diameter of the outer cylinder is $\sim$ 10 cm ($\sim$ 4 inches) and is 75 cm (29.5 inches) long.

The inner cylinder which contains the lithium absorber is sized to provide an annular void between the two cylinders. A thin-walled cylindrical baffle with a spherical nose piece containing a central hole is installed between the two

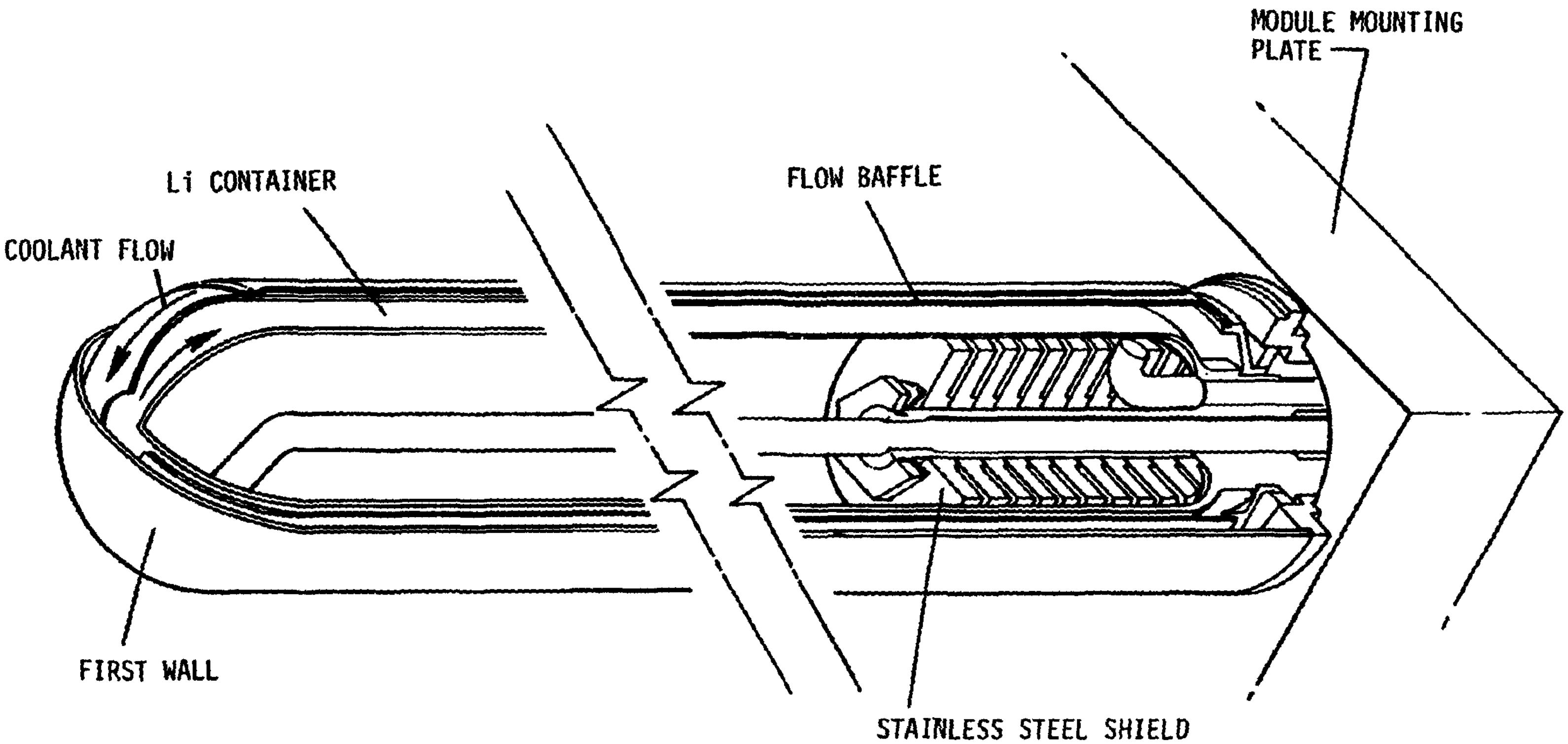

Figure 3.5-1. Cylindrical Blanket Module

cylinders. This baffle serves to create a double flow path for the coolant. The coolant flows along the inside of the outer cylinder between the cylinder and baffle to cool the outer cylinder and first wall. It then flows through the baffle central hole in the nose section and back between the inside of the baffle and the outside of the inner cylinder, cooling the inner cylinder and the lithium contained within. A detailed discussion of the concept is presented in Section 4.0.

As described, this design concept can be categorized as a hybrid concept, although if design considerations later dictate that the lithium be pressurized, it would fall into the pressurized category. The double wall feature of this concept addresses the concern for efficiently cooling the walls within acceptable pumping limits. This feature further addresses the potential problem of helium generation on heat transfer since the wall subjected to the highest heat flux is directly cooled. Because the module is a combination of cylindrical and spherical shapes, its configuration is compatible with high pressure capability. Either cylinder is capable of withstanding the total coolant pressure. Since the modules are relatively small, magnetically induced loads will be minimized.

In support of the cylindrical design concept, initial scoping analyses were performed to determine whether the design is compatible with performance requirements. The following preliminary results were obtained:

- Thermal analysis indicated that cooling of the module with $\sim$ 2% pumping power can be achieved.
- Stresses were acceptable for the Δ T required to accommodate the 1 MW/m^2 first wall particle and radiation heat flux and the coolant pressure.
- A tritium breeding ratio of 1.1 was indicated by a 1-D calculation.

Based on the above considerations, the cylindrical design was adopted for further development since the design basically conformed to the selection criteria and key design requirements. The results of additional design and analysis performed during the concept development are presented in Section 4.0.

4.0 REFERENCE CONCEPT DEVELOPMENT

This section presents a description of the design concept along with the supporting thermal and structural analyses.

4.1 DESIGN AND PERFORMANCE PARAMETERS

Table 4.1-1 lists the performance and design guidelines for the concept developed in the study. Some of the earlier guidelines were modified to be compatible with a power cycle to be used with the helium blanket coolant. Since the temperature of the helium which exits the blanket is a strong function of the first wall temperature limits, the first wall temperature guideline was increased from 400° C to 450° C. In addition, to achieve reasonable thermal performance in the power cycle, pinch point temperature considerations led to establishing an inlet coolant flow to the module of 200° C. When higher module inlet temperatures were considered to achieve the desired power conversion performance, the requirement to pump more coolant through the module to operate the structures within the temperature limits provided the incentive to change the pumping power guidelines from < 2%, as presented in Table 3.1-1, to $\leq$ 2-2.5%.

4.2 DESIGN OF REFERENCE CONCEPT

The reference cylindrical module which was recommended as a result of the concept evaluation and selection process was further developed in greater design detail. The design development process addressed module cooling, lithium containment, and module packaging to reduce the amount of metal and void fraction in order to provide for efficient tritium breeding. Requirements for piping of the auxiliary helium cooling and lithium circulating systems were considered as the basic module was integrated into the overall blanket assembly with proper consideration for structural support, vacuum boundary, maintenance and fabrication. The description of the blanket module which was developed and its integration into the blanket assembly are fully documented in Reference 13 and briefly summarized in the paragraphs which follow. Design and performance characteristics for the module concept are summarized in Table 4.2-1.

TABLE 4.1-1

PERFORMANCE AND DESIGN GUIDELINES FOR CONCEPT DEVELOPMENT

Breeding Medium	Lithium
Structural Material	Type 316 Austenitic Stainless Steel
Coolant	Pressurized Helium
Structural Concept	Modular
Vacuum Enclosure	External to Blanket
Structural Material Temperature Limits	∿ 450° C, First Wall (Radiation Damage Zone) ∿ 500° C, Low Radiation Zone ∿ 550° C, Maximum Nonstructural
Neutron Wall Loading	2-4 MW/m^2
First Wall Particle Heat Flux	0.5-1.0 MW/m^2 (without divertor)
Coolant Outlet Temperature Limit	High as practical, consistent with meeting material structural limits.
Pumping Power, $\left(\frac{\text{Pump Work}}{\text{Thermal Output}}\right)$	$\leq$ 2-2.5%
Tritium Breeding Ratio	1.2
Duty Cycle	20 minute Cycle, 95% Duty
Operating Mode	Pulsed to 10^5 Cycles
Plasma Shape	D
Minor Radius	∿ 1.5 m
Elongation	∿ 1.6
Aspect Ratio	4
Clearance (Plasma to TF Coil)	1-1.5 m
Number of TF Coils	16
Magnetic Field	TF, 11 T maximum at winding; PF, Perpendicular to TF with value of 10% of TF field.
First Wall	First wall to be part of blanket; no separate vacuum vessel inside of blanket.
Tritium Recovery	Design for recovery without removing module.

TABLE 4.2-1

CYLINDRICAL MODULE CHARACTERISTICS

Design Characteristics	
Module	∿ 10 cm O.D. x 75 cm Long
Material	Type 316 Stainless Steel
Material Thickness	
Outer Cylinder	0.16 cm (0.062 in.)
Inner Cylinder	0.16 cm (0.062 in.)
Flow Baffle	Double Thickness, 0.038 cm (0.015 in.) each
Breeding Material	Liquid Lithium
Coolant	Pressurized Helium
Performance Characteristics	
Module Thermal Power	45.1 kW/Module
Coolant Pressure	54.4 atm
Neutron Wall Loading	4 MW/m^2
First Wall Particle Heat Flux	1 MW/m^2
Coolant Temperature T_{in}, T_{out} (°C)	200, 435
Coolant Flow	35 g/s
Flow Channel Gaps	
Inlet Pass (Variable)	0.127 cm (0.05 in)-0.076 cm (0.030 in.)
Outlet Pass (Constant)	0.254 cm (0.10 in)
Material Temp. (°C)	
Outer Cylinder (First Wall)	452
Inner Cylinder	492
Lithium Max, Min	627, 461
Tritium Breeding Ratio	> 1.1
Pumping Power, $\left(\frac{\text{Pump Work}}{\text{Thermal Output}}\right)$	2.2%

4.2.1 DESIGN DESCRIPTION

4.2.1.1 BLANKET MODULE

The cylindrical module, Figures 3.5-1 and 4.2-1, consists of a double wall stainless steel cylinder with a spherical end designed for a helium coolant pressure of approximately 5.5 MPa (54 atm). The outside diameter of the outer cylinder is approximately 10.2 cm (4 inches) and it is 75 cm (29.5 inches) long.

The inner cylinder, which contains the breeding media, is sized such that an annular void is created between the two cylinders analogous to a Thermos bottle. The wall of each cylinder including the spherical ends is approximately 1.6 mm (0.062 inches) thick. Installed between the two cylinders is a third stainless steel thin-walled cylindrical baffle with a spherical nose piece. This thin-walled baffle, approximately 0.38 mm (0.015 inches) thick, has a 2.54 cm (1.0 inch) diameter hole in the center of the spherical end and dimpled projections located on the cylindrical surface of the baffle. The dimpled projections are raised on the inner surface of the baffle only. Another thin-walled cylinder with dimpled projections on the inner and outer surfaces is fitted over the cylindrical portion of the thin-walled baffle and sized to provide a dead space for stagnant coolant gas between the thin-walled baffle cylinder assembly. The internal and external dimples on the baffle assembly provide concentric centering of the inner and outer cylinders with the baffle assembly sandwiched in between. The 2.54 cm (1.0 inch) diameter hole in the spherical end of the baffle serves as a port to connect the passage between the outer cylinder and baffle to the passage between the baffle and the inner cylinder.

The back end (end farthest from the plasma) of the inner cylinder is welded to a funnel-shaped hub. The funnel-shaped hub contains an inlet and an outlet port for the liquid lithium and a central outlet passage for the helium coolant. Prior to welding the inner cylinder to the funnel-shaped hub, appropriately shaped and oriented inlet and outlet tubes are welded to their respective lithium ports. The location and orientation of the blanket module will determine the positioning of these tubes.

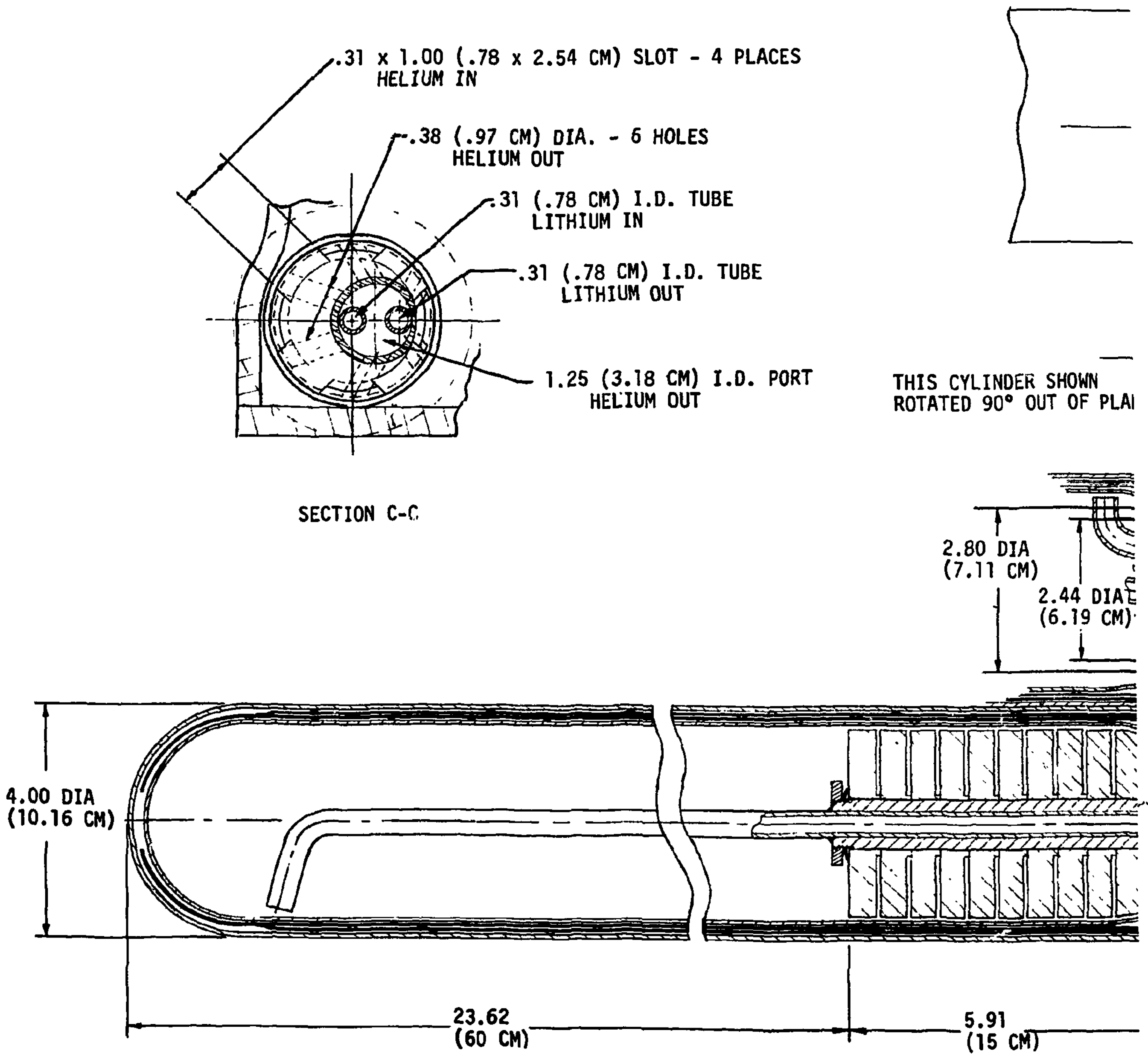

Figure 4.2-1. Cylindrical

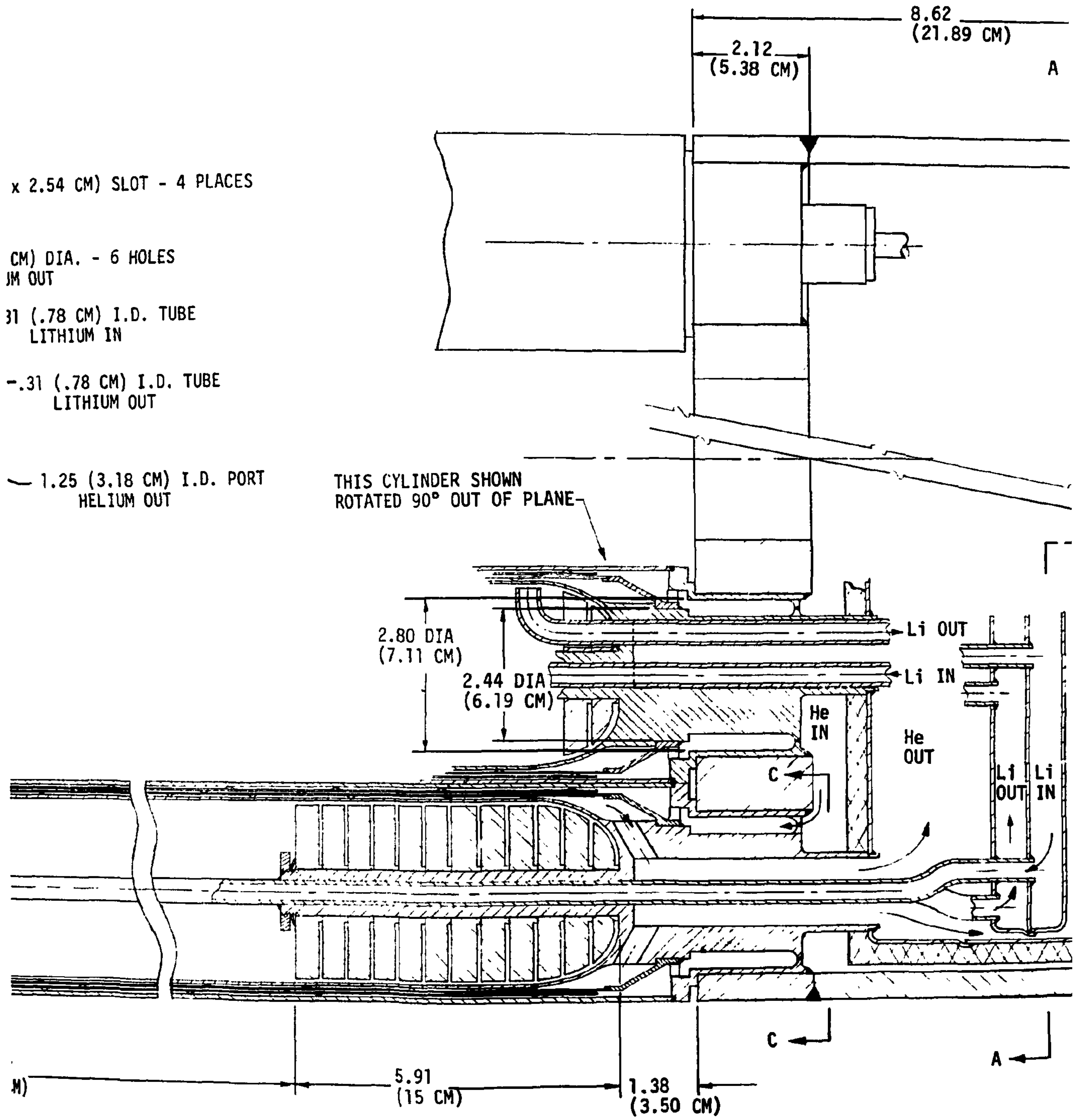

Figure 4.2-1. Cylindrical Module-Concept Design

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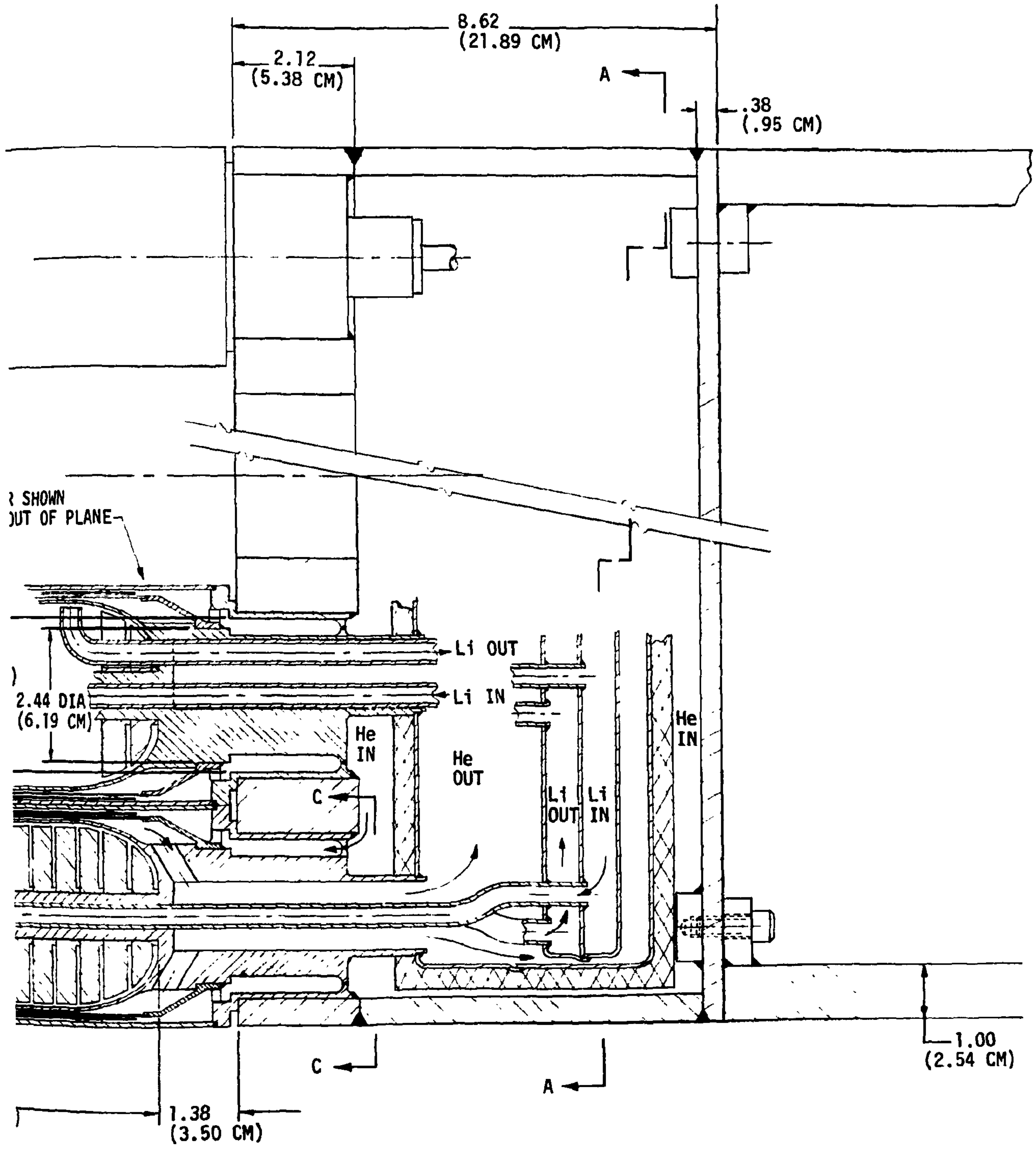

ndrical Module Concept Design

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Several thick washer-shaped shields with integral spacers, made of stainless steel, are fitted to the center post of the lithium cylinder hub. The thick washers provide shielding and reflection of neutrons to enhance breeding and provide shielding for the rear structural members of the module, while the integral spacers provide paths for circulation of the liquid lithium.

After assembly of the lithium cylinder, the unit is installed in the thin-walled baffle. A deep dished washer-shaped hub is then assembled over the hub of the lithium cylinder and is welded to the mating end of the baffle cylinder. The hub assembly of the inner cylinder and baffle is attached to a circular manifold by a circular omega seal weld located at the rear of the circular manifold. Finally, the outer cylinder is fitted over the inner cylinder and baffle assembly and the open end is welded to a flange section at the front end of the circular manifold.

In tracing the helium coolant path, Figure 4.2-1, the pressurized helium enters the circular manifold through four slots in the circular omega seal region at the rear of the circular manifold. The coolant enters an annular plenum chamber and proceeds along the annular void formed by the baffle and the outer cylinder. As the helium coolant travels at relatively low velocity along the outer annulus to the spherical end of the module, it removes heat from the outer cylinder.

Once the helium reaches the spherical end, it passes through the 2.54 cm (1 inch) diameter hole in the end of the baffle and returns between the baffle and the inner cylinder, picking up heat that has been transmitted to the walls of the inner cylinder by the lithium. The heated helium coolant then enters the helium outlet plenum at the rear of the inner cylinder and exits to a central port in the hub of the inner cylinder through six radially positioned circular passages.

To keep the void fraction to a minimum in this design, the cylindrical blanket modules are attached to a blanket module mounting plate using a triangular pitch configuration. The average void fraction between modules using this concept is approximately 17%. The blanket modules are fitted and welded into 7.62 cm (3.0 inch) diameter holes located on a triangular pitch of 10.2 cm (4.0 inches). The mounting plates are sized to contain three rows of cylindrical blanket modules and vary in horizontal length from 111.8 cm (44 inches) to 50.8 cm (20 inches) depending on the location within the reactor. The cylindrical blanket module subassemblies are self-contained in that each subassembly has its own plenum chamber and collector manifolds for the lithium and coolant gas, Figures 4.2-1 and 4.2-2.

4.2.1.2 SUPPORT STRUCTURE

Forty-eight D-shaped structure segments are required for mounting the subassemblies. Each structure is tapered such that when the 48 segments are assembled they form a solid torus. Figure 4.2-2 shows an assembly of 6 segments and Figure 4.2-3 shows the relationship of the blanket sector assemblies in a toroidal configuration within the TF coils. The structure is fabricated using five straight sections to form the outer leg of the D. The blanket subassemblies are bolted to the inner peripheral surface of the D-shaped structure with the spherical nose of each cylindrical module pointing to the plasma. In addition to supporting the blanket subassemblies, the D-shaped structure houses the piping headers for supplying helium and lithium to the individual subassemblies. Each blanket structure segment assembly (less shielding) weighs approximately 38 tonnes, including the weight of the lithium.

Every third blanket structural segment located midway between adjacent TF coils is supported on a pedestal projecting from the foundation, Figure 4.2-4. Two pairs of brackets are mounted on each side of these pedestals and are interconnected by a pair of short beam-type structures. The blanket structural segments located on either side of the blanket structure segment mounted on the pedestals are supported by these pairs of short beams. As the torus formed by the 48 blanket structural segments grows circumferentially due to thermal expansion, each segment will move radially. This motion is accommodated by a roller assembly mount attached to the base of each blanket structural segment.

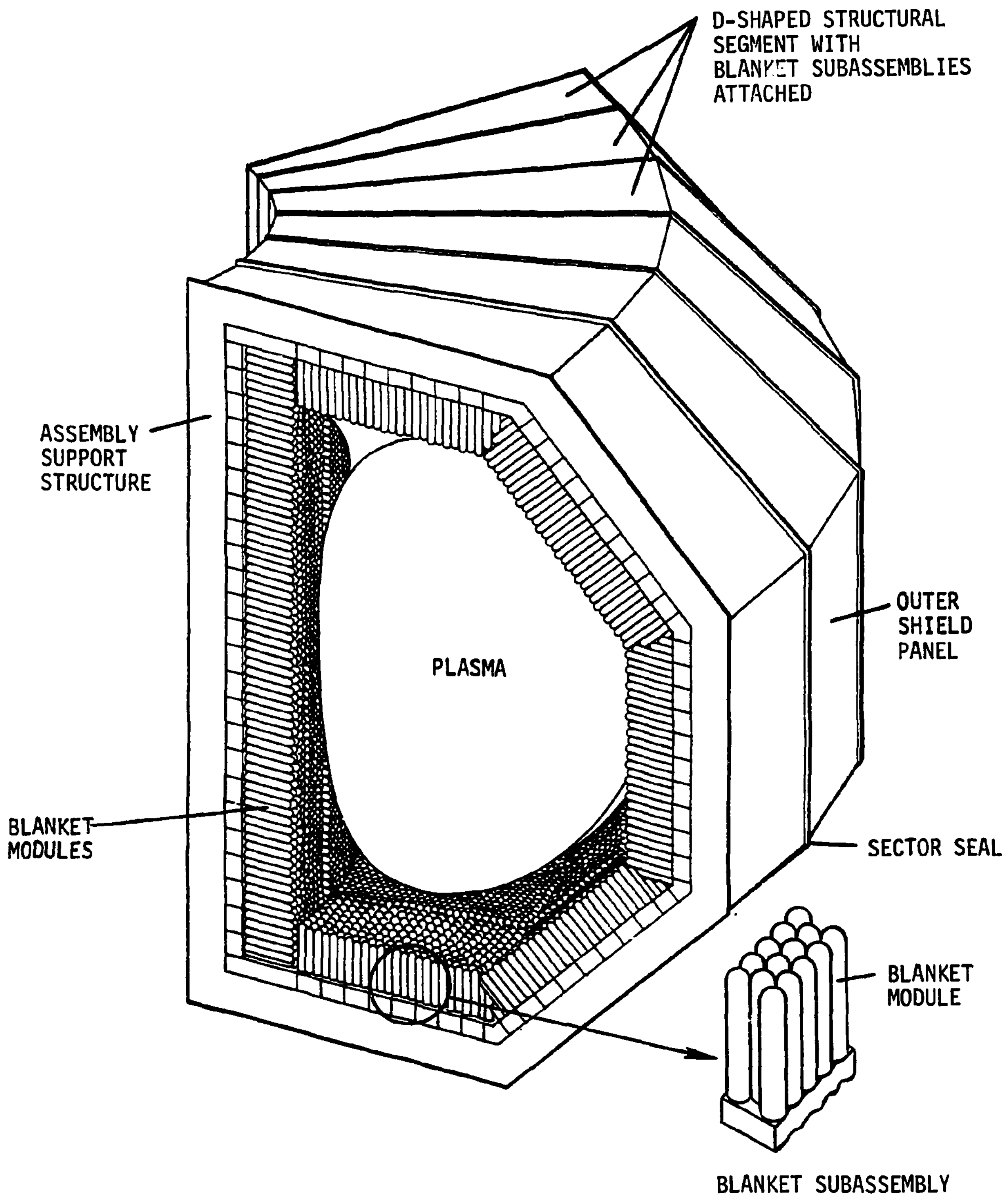


Figure 4.2-2. Blanket Sector Assembly

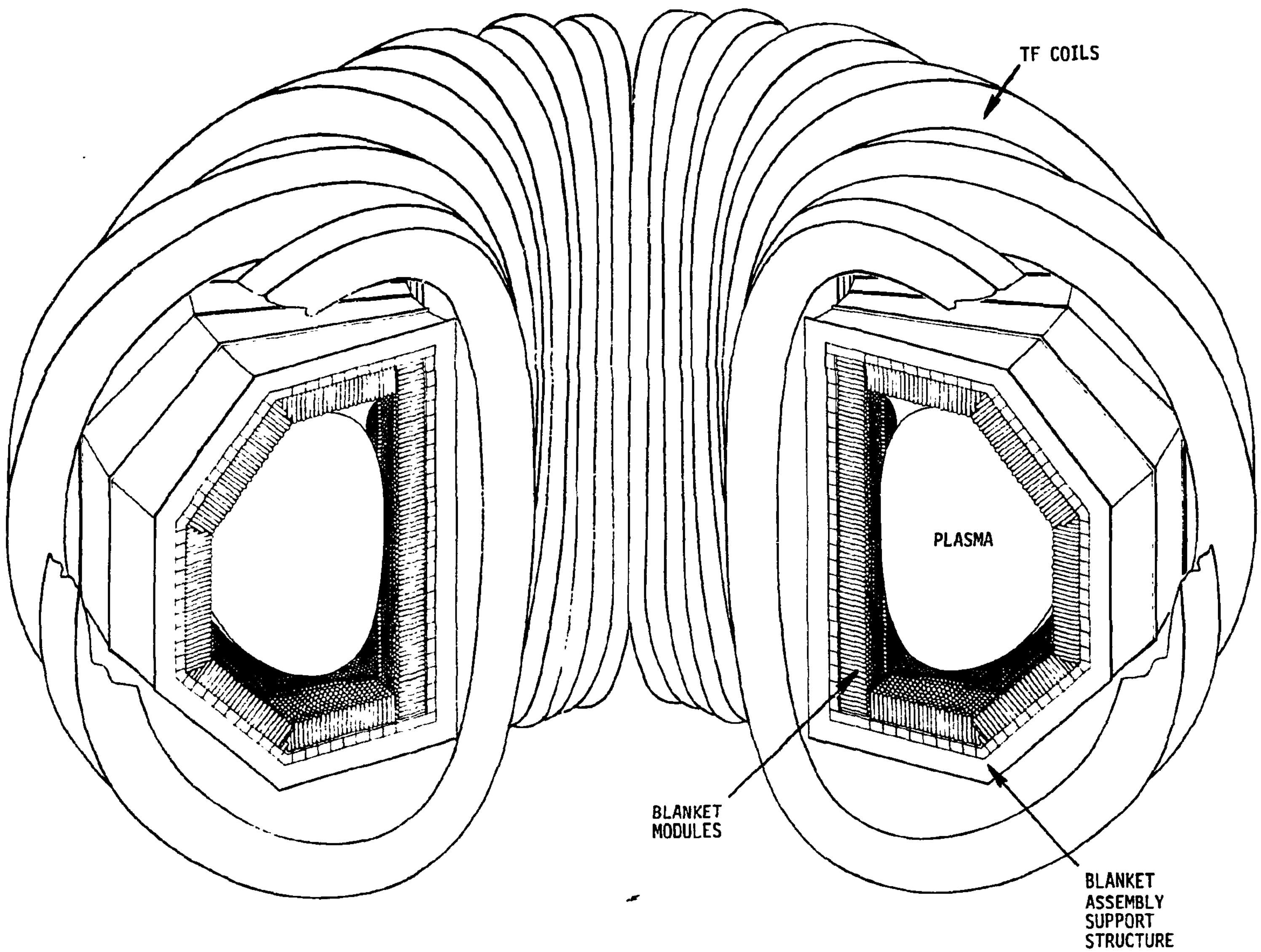


Figure 4.2-3. Toroidal Field Coil/Blanket Arrangement

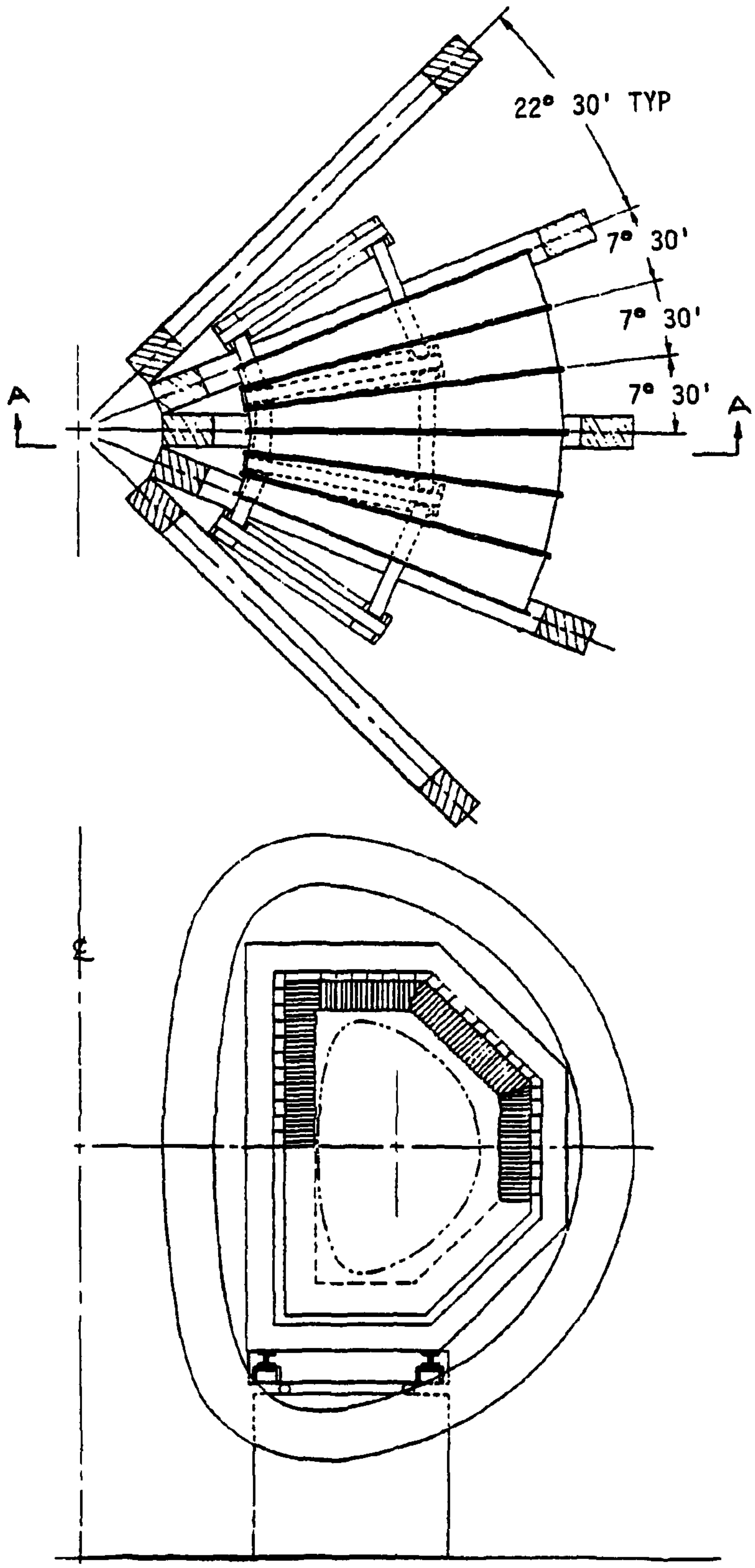


Figure 4.2-4. Blanket Module Mounting

4.2.1.3 INTERFACING AUXILIARY SYSTEMS

Interfacing systems such as piping, headers, and shielding were not developed to any appreciable detail in the FY 78 program. However, because these systems have a direct influence on the design of the blanket modules, consideration was given to their space requirements and location. The peripheral area of the D-shaped structure is reserved for the peripheral headers and associated piping. The remainder of the area contains shielding. In designing the outer peripheral closure for the D-shaped structure, the concept of using borated water with steel balls as the possible shielding medium was considered.

Although, as stated above, the piping and header system concepts were not developed, a possible arrangement shown pictorially in Figure 4.2-5 might be considered. The peripheral inlet and outlet headers, as envisioned, circumvent the D-shaped structure and are divided into four lengths. Each length starts at the midplane of the D-shaped structure with two headers terminating at the top of the structure and two at the bottom. This arrangement limits the servicing of one-fourth of the blanket subassemblies to each header. These headers are connected to larger toroidal manifolds located on the top and bottom of the reactor.

4.2.2 RESPONSE OF DESIGN TO KEY REQUIREMENTS

In the process of developing the design concept, the key considerations of first wall cooling, pressurization of the module as a result of a postulated coolant leak, and magnetic loads on the module identified in Section 3.4 were addressed. In addition, the important items such as assembly and maintenance, manufacturing, and helium and tritium removal were specifically considered. The response of the design to the above is briefly summarized in this section.

4.2.2.1 WALL COOLING

To achieve integral wall cooling the reference design concept uses a double wall approach. The concentric cylindrically shaped vessels provide for the helium coolant path to effectively accomplish cooling of the outer cylinder (first wall) and cooling of the internal cylinder which contains the liquid lithium.

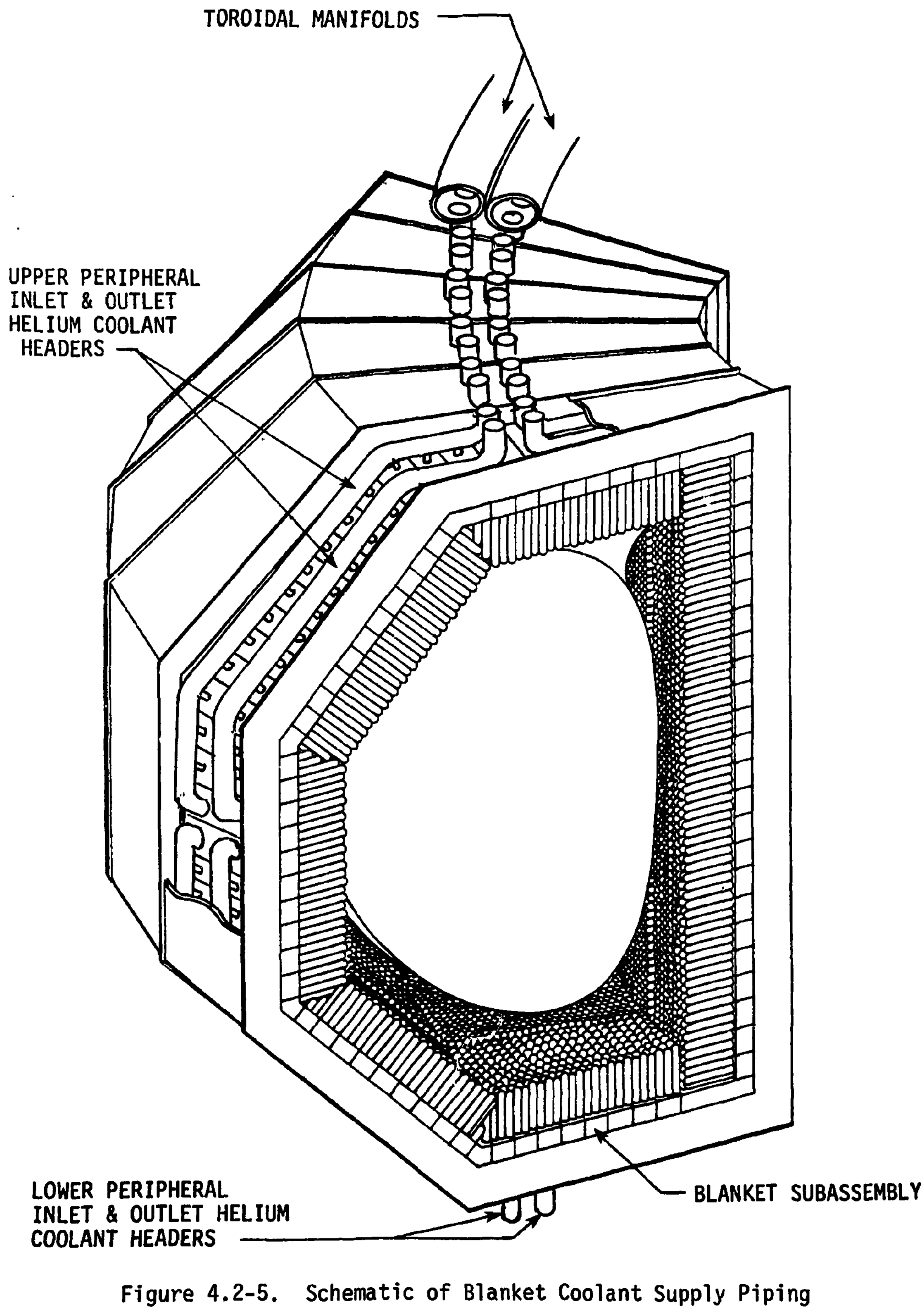


Figure 4.2-5. Schematic of Blanket Coolant Supply Piping

4.2.2.2 MODULE PRESSURIZATION

In order to prevent the leak problems associated with possible overpressurization of the lithium container, the structurally efficient cylindrical shape for the outer and inner vessels is used in the reference concept, and both vessels are designed to carry the full coolant pressure.

4.2.2.3 MAGNETIC CONSIDERATIONS

The reference design concept tends to reduce the magnetically induced forces and torques on the modules by using small diameter cylinders. In addition, the use of small diameter cylinders mounted in small subassemblies is beneficial in permitting the pulsed field to penetrate the blanket module.

4.2.2.4 ASSEMBLY AND MAINTENANCE CONSIDERATIONS

The service and maintenance philosophy behind the cylindrical blanket module concept is based on modular rather than unit assembly/disassembly operations. In practice the blanket module assembly of the reactor is made up of 48 D-shaped structural segments. To service the cylindrical blanket subassemblies attached to the D-shaped segment, the segment is first removed from the reactor. Details of the servicing procedure are discussed in Section 5.4.

4.2.2.5 MANUFACTURING CONSIDERATIONS

In the reference concept, $\sim$ 66,000 cylindrical modules are required for a complete reactor. Quantities of modules of this magnitude dictate that volume manufacturing techniques be considered as part of the module design effort.

The inner and outer cylinders of each module used in this concept have integral spherical closed end caps at one end and full diameter openings at the other end. The open ends are welded to circular hubs and flanges that, in turn, are attached to a common mounting plate. This scheme simplifies the method for assembly of cylinders within cylinders. The mounting plate is made a part of the helium plenum chamber by welding it to a structural stainless steel plate enclosure. These major components are amenable to mass production and assembly techniques using present state-of-the-art machines and procedures. The blanket subassembly manifolds which interconnect the individual modules are made in two parts for ease of assembly and disassembly. The two-part manifold also provides maximum access to the individual module lithium supply tubes for automatic welding and inspection to assure optimum reliability. Omega type seal welds are used wherever feasible throughout the design to permit the use of present state-of-the-art circular welding and cutting machines.

4.2.2.6 HELIUM AND TRITIUM REMOVAL

The helium and tritium flow path through the interior of the cylindrical module was described in Section 4.2.1.1 and is shown in Figure 4.2-1. The helium and tritium removal discussed in this section is restricted to the inner cylinder of the module which contains the liquid lithium. In the discussion of wall cooling (Section 3.4.1), it was noted that helium removal is required, since helium gas is generated in sufficient quantities to accumulate in the lithium.

The present design concept requires that the inlet and outlet tubes for transporting the lithium be bent and positioned such that the opening of the inlet tube is located at the lowest point and the outlet tube opening is located at the highest point. Specific location of the tubes is necessary to assure proper circulation and recovery of the lithium and to remove helium gas pockets within the cylinder during the pumping process. Pumping of the liquid lithium is done when the magnetically induced loads on the liquid lithium are at a minimum.

4.2.3 ADVANTAGES AND DISADVANTAGES OF THE REFERENCE CONCEPT

The reference cylindrical blanket module concept was developed to resolve some of the problem areas that surfaced as a result of the evaluation of the candidate blanket modules.

Some of the advantages of this design are:

- The use of cylindrical modules resolves the structural design problems associated with larger blanket modules which employed rectangular shapes.

- Direct cooling of the first wall protects the walls from overheating because of the lithium-free surface caused by helium generation, and the concentric cylinder arrangement provides a double barrier between the lithium and the plasma.

- The small module concept appears to minimize the forces on the lithium container resulting from the magnetic loads when the field is pulsed.

- The module is designed with all weld joints located near the rear so that no highly stressed weld is in a highly irradiated area.

- The simple efficient module shape is desirable from the standpoint of fabrication by mass production techniques.

- The simple flow arrangement provides an effective cooling method which can be calculated and readily verified in a configuration readily adaptable to component testing.

- The weld geometry is adaptable to automatic weld machine processes for a more reliable and automated production.

- Minimum metal contact between the hub and the circular manifold on the outer cylinder is achieved by use of a thin membrane seal. This reduces the heat transfer to the incoming cold helium and the blanket module mounting plate.

Some of the disadvantages are:

- Design of the modules with capability to withstand the stresses from the coolant pressure results in many smaller modules which may require more physical length of piping and additional connections to the coolant supply system.

- The voids between the cylinders result in longer modules in order to obtain the desired effective lithium breeding volume.

- The large number of modules may impact the blanket system operating reliability and require more quality control inspection, checkout, and time to assure that reliable components and assemblies are fabricated.

4.2.4 AREAS FOR DESIGN IMPROVEMENT

Helium generation in the lithium is a potential problem in that the gas, if not adequately removed, may collect in certain areas on the inner surface of the lithium container and affect the local heat transfer from the lithium. A second problem area associated with the lithium container is the effect of magnetic fields on the blanket modules. The size of the subassemblies as affected by eddy currents will need further investigation. The potential for increasing the module size should be investigated to reduce the quantity and enhance reliability.

Various stacking patterns for the cylindrical modules should be investigated to assure optimal designs to: (1) minimize the void fraction and the metal/lithium volume ratio, (2) assure maximum vacuum pumping efficiencies by avoiding areas that are difficult to evacuate, and (3) reduce the amount of manifolding and piping by effective arrangement of the modules and subassemblies to enhance the system reliability.

4.3 THERMAL PERFORMANCE OF BLANKET MODULE

4.3.1 PURPOSE AND SCOPE

The thermal and hydraulic performance of the reference module design described in Section 4.2 is analyzed and fully documented in Reference 14. The results are summarized in this section. The purposes of the analysis are to determine and assess the thermal performance to support the reference design concept and to select a reference operating point compatible with a reasonable power conversion system.

The scope of the thermal study consists of analyzing the reference design. In-depth optimization of the design to obtain the best performance was not performed. Steady-state and transient thermal conditions of the module were determined to provide input for structural analysis.

4.3.2 DESIGN GUIDELINES AND REQUIREMENTS

The blanket module design guidelines and thermal performance requirements are discussed in Section 4.1 and tabulated in Table 4.1-1. The key thermal design parameters are the following:

- Wall loading: 4 MW/m^2 of neutron wall loading and 1 MW/m^2 of particle surface heat flux.
- Structural material temperature limits: $\sim$ 450° C on the first wall and $\sim$ 500° C at the low stressed structure.
- Pumping power requirement: $\leq$ 2.0-2.5% of blanket thermal power.
- Coolant outlet temperature: as high as practical.
- Lithium temperature range: 180° C-1310° C (melting to boiling point).

4.3.3 THERMAL ANALYSIS

4.3.3.1 METHOD OF ANALYSIS

The thermal-hydraulic analysis of the module was performed by utilizing a computer code for solving the multidimensional heat conduction equation for the solid materials and the conservation equations for the fluid in the cooling channels. Multidimensional heat transfer between nodes is modeled. The fluid flow analysis is, however, one-dimensional in the direction of the flow of the coolant in the channel.

4.3.3.2 MODEL OF ANALYSIS

Because of axial symmetry of the module design, the model considers the axial and radial heat transfer in the solid materials of the module. The coolant enters the outer gap channel at the base of the module, turns around at the tip of the hemisphere, and exits from the inner gap channel also at the base of the module. In order to reduce heat exchange between the counterflowing coolant, the baffle must provide insulation between the flow streams. A stagnant helium gap of 0.23 cm between the inner and the outer baffle walls is sufficient. This gap, extending along the entire length of the baffle, is included in the model for the analyses.

The incident surface heat flux acting on the spherical surface was considered to follow a cosine function variation with the peak value at the top center of the hemisphere.

The sizes of the outer and the inner coolant flow gaps of the reference design are 0.127 cm and 0.254 cm, respectively, except along the curved portion of the outer flow gap at the nose of the module. The outer baffle along the curvature is made thicker, so that the flow gap is reduced as the flow approaches the central hole at the nose. Tapering of the gap size further reduces the flow area in the curved and convergent region of the dome. The heat transfer coefficient on the inner surface of the front wall is thereby significantly increased to reduce the front wall material temperature.

The pressure loss at the exit of the outer flow gap was calculated to be approximately the velocity head and that at the entrance to the inner gap was calculated to be about 0.8 velocity head. The calculations were based on the losses due to two 90° turns and the associated flow area changes. The heat transfer coefficient at the center of the dome was assumed to be 150% of that at the exit due to the turbulence at the turnaround. Test data by Boelter et al.[15] using air in turbulent flow indicated that the local Nusselt number at the bend in a 90° turn near the entry region of a tube is about twice the Nusselt number away from the bend, indicating heat transfer enhancement on the order of two. However, since the flow at the turnaround is not exactly simulated in the experiments, the heat transfer coefficient was assumed to be improved by a factor of 1.5 instead of 2.

4.3.4 ANALYTICAL RESULTS

4.3.4.1 STEADY-STATE PERFORMANCE CURVES

Because of the high duty cycle (95%) and the thin cylinder walls, the thermal conditions of the module reach steady state quickly during each pulse. Steady-state analyses were carried out to determine the module thermal conditions. The performance curves of the module design with a helium inlet pressure of 5.5 MPa (54.4 atm) and inlet temperature of 200° C are shown in Figure 4.3-1.

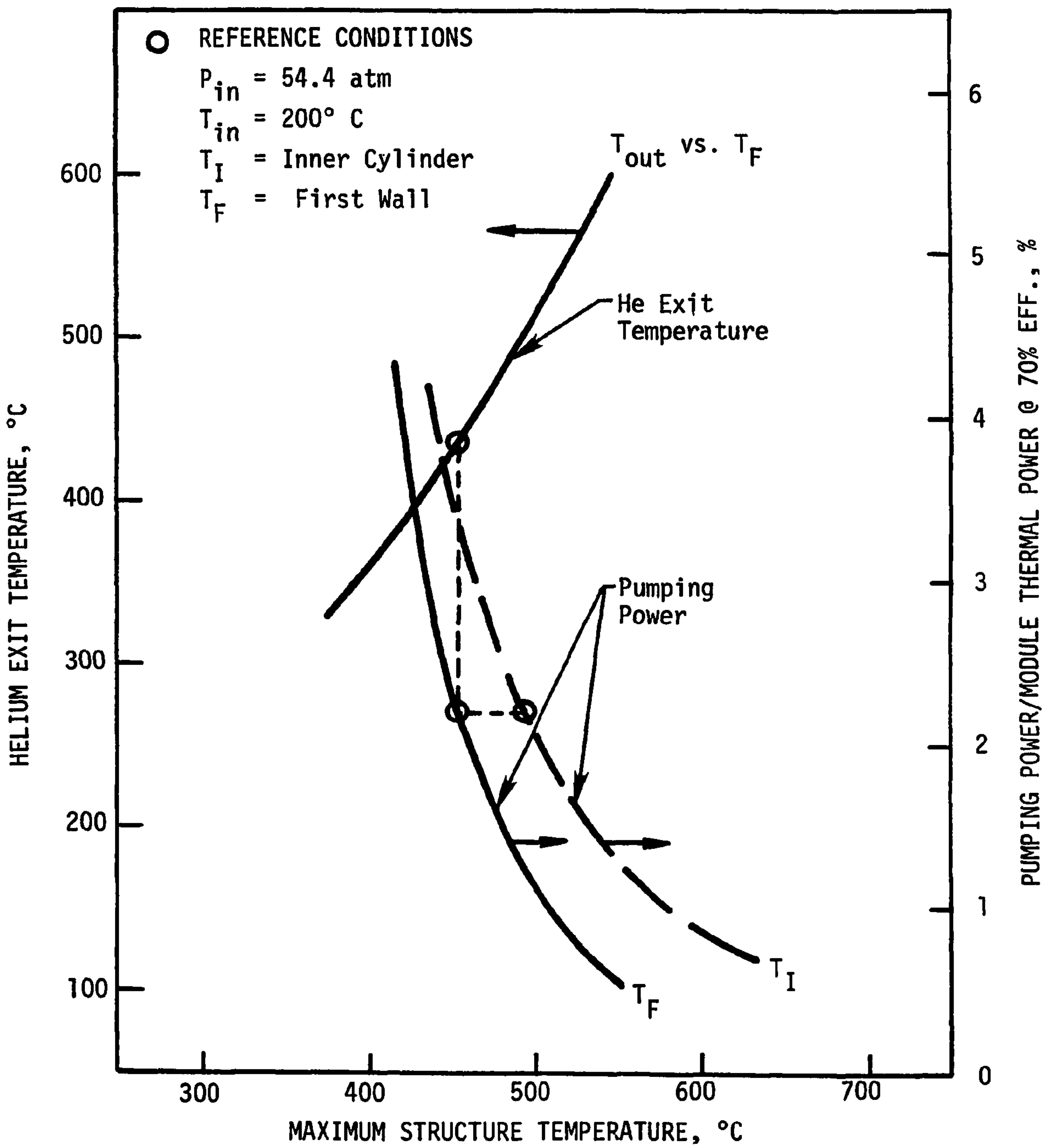


Figure 4.3-1. Steady-state Operating Curves for the Blanket Module

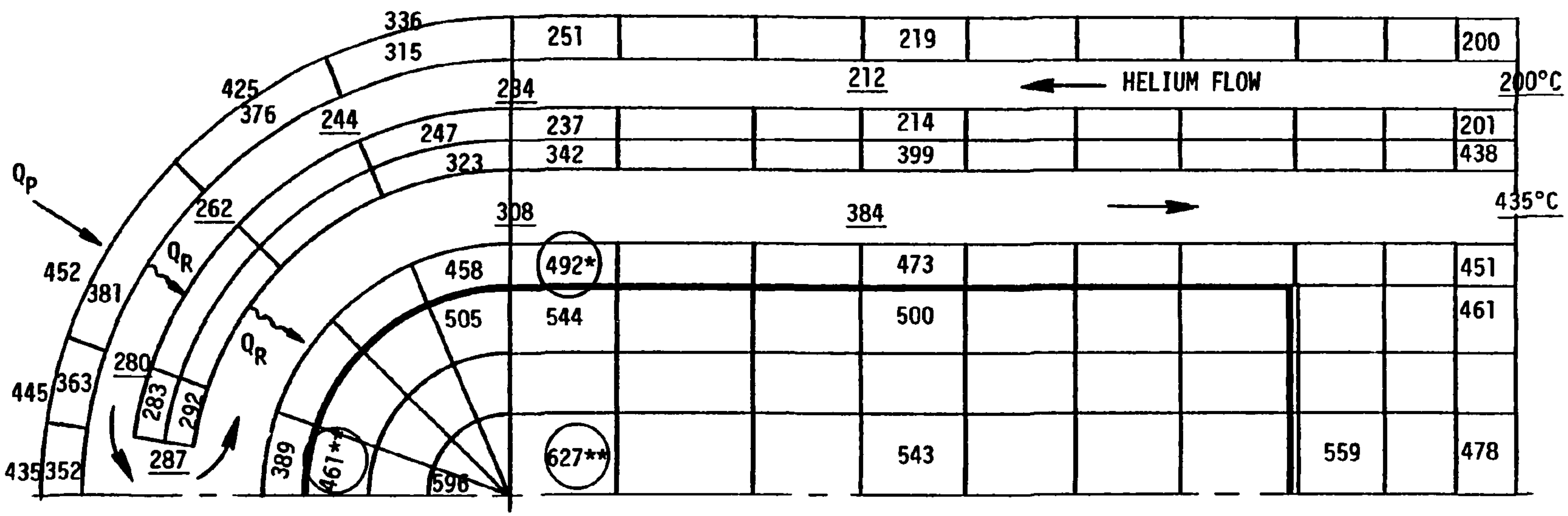
The maximum inner and outer shell (first wall) temperatures are plotted versus the required pumping power and the helium coolant exit temperature is plotted versus the shell first wall temperature. The curves show that the required pumping power is very sensitive to the shell material temperatures with a helium coolant inlet temperature of 200° C. The allowable operating region is to the left of 500° C on the inner shell temperature curve or to the left of 450° C on the front wall temperature curve. The maximum attainable helium exit temperature in this design is about 430° C because of the first wall structure material temperature limits. The reference operating conditions were selected at the points shown by the circles on the curves.

4.3.4.2 REFERENCE OPERATING CONDITIONS

From the performance curves shown in Figure 4.3-1 and the design requirements given in Section 4.3.2, the reference operating point is selected at the points shown by the circles in Figure 4.3-1. The steady-state reference thermal conditions are as follows: maximum inner cylinder material temperature is 492° C, maximum first wall temperature is 452° C, helium coolant exit temperature is 435° C, and the required pumping power at an assumed 70% pump efficiency is 2.2%.

The spatial temperature distributions within the module are shown in Figure 4.3-2. The figure also shows the model for the analysis. In this figure the temperatures are shown at some of the key locations, although the temperatures at every node of the model were calculated. It is shown in the figure that the maximum front wall temperature did not occur at the center of the dome, but at about 35° from the module central axis. This is due to the increasing coolant velocity, as the coolant flows toward the nose of the hemisphere, and the decreasing surface heat flux along the curvature of the dome.

The maximum inner cylinder temperature occurs at the transition point from the inner dome to the straight portion of the inner cylinder. The maximum and minimum lithium temperatures are 627° C and 461° C, respectively, and are within the temperature range to maintain the lithium in a molten state. The



Helium Inlet Pressure = 54.4 atm (800 psia)

Helium Inlet Temperature = 200° C

Neutron Wall Loading = 4 MW/m^2

* Max. SS Structure Temp.

** Min. & Max. Li Temp.

Ideal Insulation @ Baffle

xx = Material Temp, °C

xx = Coolant Temp, °C

Figure 4.3-2. Steady-state Material and Coolant Temperatures of Module maximum stainless steel shield and reflector plate temperature as shown in the figure is 559° C. However, these are not structural members, and these temperatures are considered acceptable.

The axial helium coolant and cylindrical shell temperature distributions are shown in Figure 4.3-3. It is seen that about 37% of the helium coolant temperature rise occurs at the outer gap (between the flow baffle and outer cylinder) and the remaining 63% occurs at the inner gap (between the flow baffle and inner cylinder). This ratio is achieved by thermally isolating the inner and outer flow paths.

4.3.4.3 POWER CONVERSION CYCLE CONSIDERATIONS

One of the thermal design requirements of the blanket is to obtain a coolant exit temperature as high as practical for a reasonable power conversion efficiency. The maximum coolant exit temperature attainable with this module design as shown by the performance curves in Figure 4.3-1 is approximately 440° C. This exit temperature is constrained by the structural material limits. With a coolant exit temperature of 435° C and an inlet temperature of 200° C, a conventional steam turbine-generator cycle can be considered. A feasible one is a single-pressure, one-extraction, regenerative cycle with a feedwater temperature of 93° C (200° F). The cycle diagram and the temperature-enthalpy plot of the steam generator portion of the cycle are shown in Figure 4.3-4.

The temperature difference at the pinch point (b) was selected to be 20° C and that at the steam generator exit, 15° C. With these temperatures, the highest steam pressure that can be utilized in the cycle is 2.41 MPa (350 psi). The extraction point is at a pressure of 69 kPa (10 psia). Assuming a turbine stage efficiency of 80% before the extraction point and 75% after the extraction point, the cycle thermal efficiency was calculated to be about 30.8%, which is considered reasonable for this study. With the existing helium temperatures and pressures, a direct helium gas turbine cycle for power conversion provides an unreasonable thermal efficiency of less than 20%.

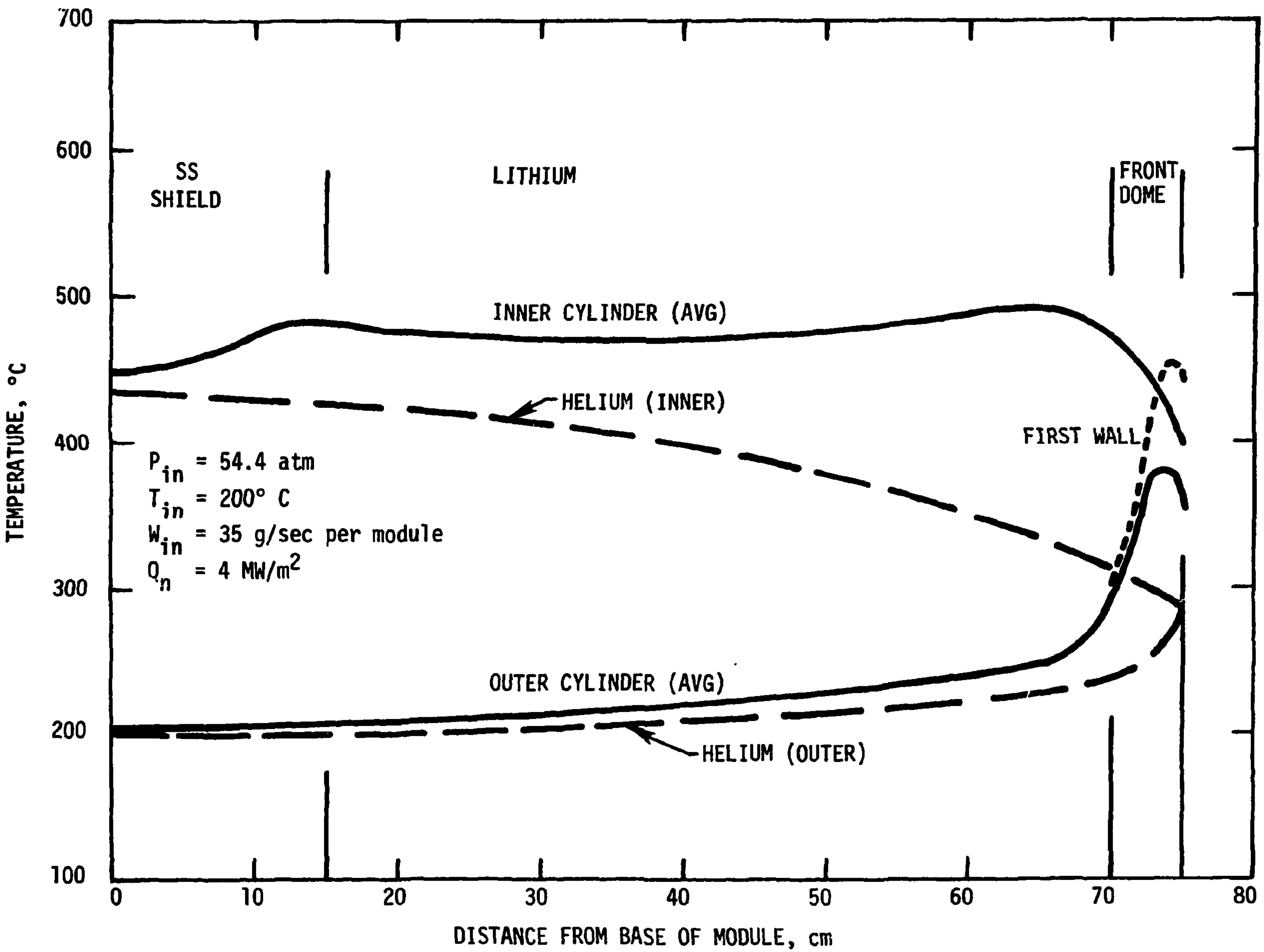


Figure 4.3-3. Axial Temperature Profiles of Helium and Structural Cylinders

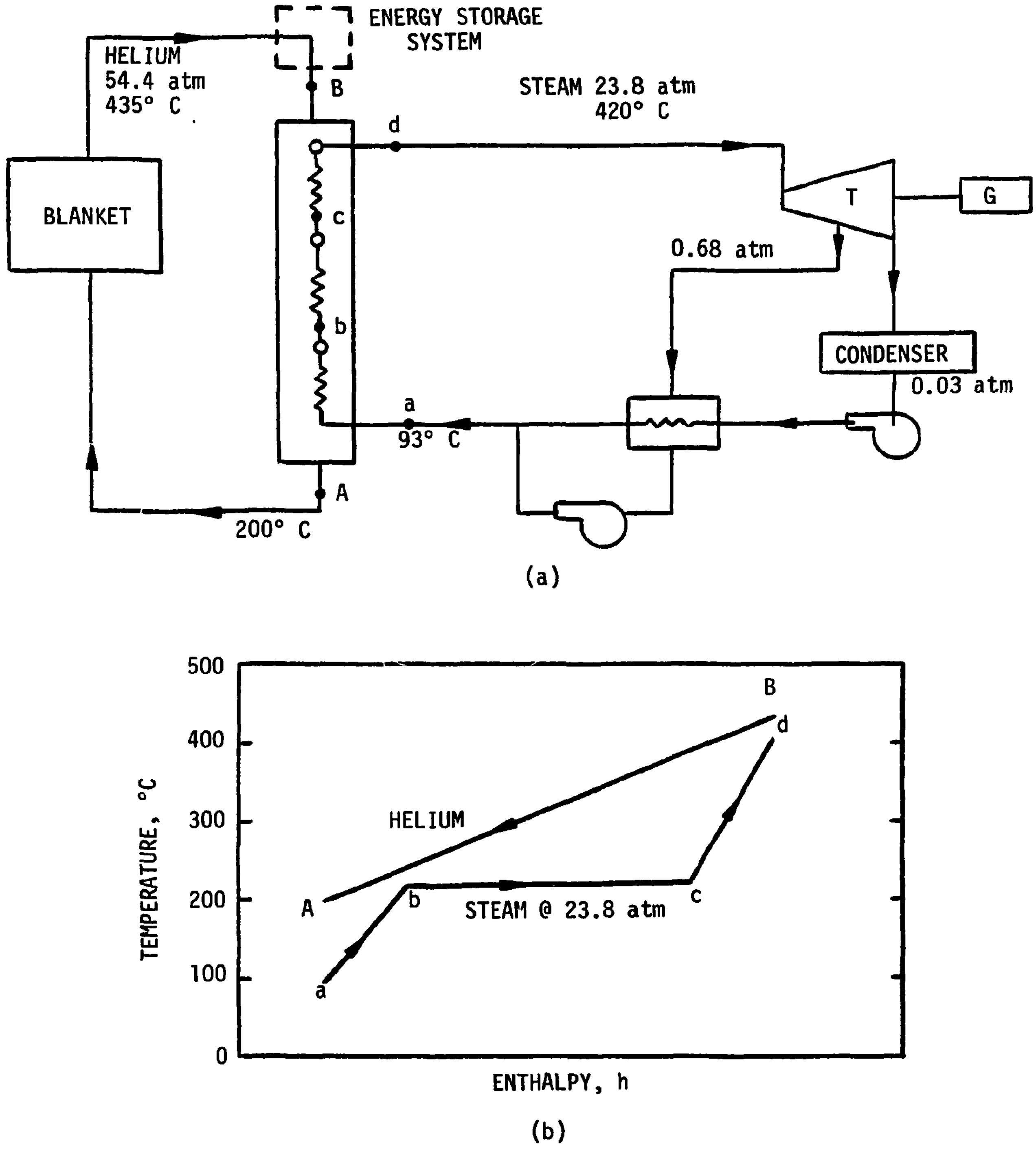


Figure 4.3-4. Helium/Steam Power Conversion Cycle Configuration

4.3.4.4 TRANSIENT CHARACTERISTICS

The transient thermal conditions of the module during a plasma-off of 60 seconds and during the subsequent plasma-on are shown in Figure 4.3-5. The values at the negative times represent the reference steady-state operating point conditions. The analysis assumed that the plasma was initiated and turned off in a stepwise manner. It is seen from the figure that the front wall temperature responds rapidly as the surface heat flux is terminated or resumed. At the end of the 60 second plasma-off period the minimum lithium temperature decreases to about 235° C which is still above the melting point. The helium exit temperature decreases, however, to about 300° C from 435° C. This coolant temperature variation during a power cycle would affect the operation of the steam generator in the power conversion system. This is a problem common to utilizing pulsed operation in power conversion. Some buffered energy storage provision is needed to minimize this effect. It is also seen from the figure that if the plasma-off period is longer than 60 seconds, the helium exit temperature would continue to drop. Since the helium inlet temperature is maintained at 200° C the minimum lithium temperature would approach 200° C in 2 to 3 minutes. The effect of the plasma-off time on the performance of the module is, therefore, quite significant.

After the plasma is resumed following the 60 second plasma-off, the thermal conditions take about 2 to 3 minutes to reach the previous steady-state conditions.

4.3.5 ADDITIONAL THERMAL ANALYSIS REQUIREMENTS

Steady-state and transient analysis have been performed for the module to determine the performance limits and to establish a set of reference operating conditions for the module design described in this report. Additional thermal analyses are required to optimize the design parameters in order to improve the thermal and structural performances of the design. The areas to be considered for optimization include the following items:

(1) Size of the module — diameter and length.

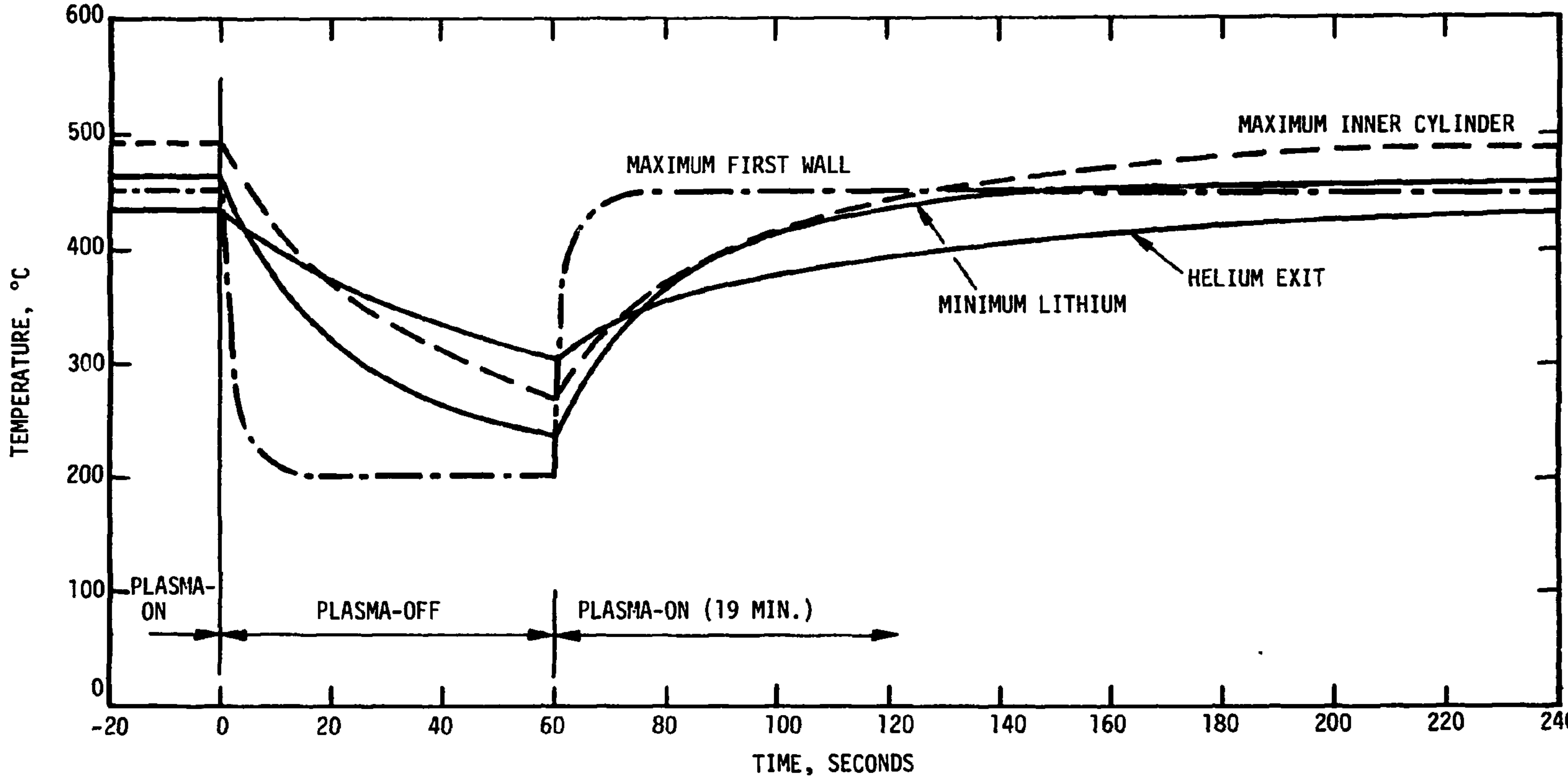


Figure 4.3-5. Transient Thermal Conditions During a Plasma On-Off Cycle

(2) Coolant channel arrangement — flow areas and different ways of achieving cooling of the module concept.

(3) Coolant pressure and temperature — for efficient use of coolant for power conversion.

(4) Power conversion cycle configuration — types of cycles and operating conditions to optimize combined module and power cycle efficiency.

4.4 STRUCTURAL ANALYSIS AND EVALUATION

4.4.1 PURPOSE AND SCOPE

The purpose of the blanket module structural evaluation was to determine the structural acceptability of the design under normal reactor operating conditions. The evaluation was performed in relation to criteria established to assess the potential for first wall failure and the resultant coolant leakage into the plasma. Normal reactor operation was considered to consist of 1×10^5 plasma on-off cycles with peak plasma-on neutron and surface heat fluxes of 4 MW/m^2 and 1 MW/m^2, respectively. The plasma was considered to be on for 19 minutes and off for 1 minute.

The present blanket module structural evaluation was performed only for the first wall because it must operate with the most severe incident surface particle heat flux in combination with high neutron irradiation. Other structural regions of the blanket module are less severely loaded and were judged not to limit the blanket module replacement schedules established by first wall considerations.

The structural analysis and evaluation consisted of analyses to determine pressure, thermal, and swelling loads on the module for normal operating conditions. Using these loads, a worst case duty cycle was established which gave stress levels to be considered for evaluating structural adequacy against defined acceptance criteria. The structural evaluation was performed to determine the relative potential for failure based on crack growth due to fatigue, brittle fracture, or excessive deformation which might lead to changes in flow channel gaps, causing hot spots leading to material degradation because of elevated operating temperatures. The following sections briefly summarize the results. A more comprehensive discussion of this analysis and evaluation is contained in Reference 16.

4.4.2 ASSUMPTIONS AND CRITERIA

In performing the structural evaluation of the blanket module first wall, the first wall hemispherical nose and cylinder were assumed to be constructed from 316-SS in a 20% cold worked (CW) condition. Further, it was assumed that the peak first wall temperature was to be maintained below 450° C in order to minimize degradation of 20% CW-316-SS mechanical properties due to neutron irradiation. With regard to material property assumptions, the current fission data base for 20% CW-316-SS in fast breeder reactor structural analysis was used in lieu of the unknown effects of fusion neutron environments on the blanket module wall.

4.4.2.1 CRITERIA

Two types of potential failures were considered in this evaluation. These were:

- Coolant leakage into the plasma which is caused by either crack growth or brittle fracture.
- Excessive deformation of the module which results in a variation of coolant channel gap leading to a potential hot spot temperature in excess of the normal peak wall temperature of 450° C.

Criteria were established such that if positive margins of safety are calculated it is concluded that the module is structurally acceptable. If the criteria are violated by negative calculated margins of safety, the module is not structurally adequate. The calculated margin of safety provides an indication of the relative potential for failure by each of the failure mechanisms. The criteria are summarized and presented in Table 4.4-1.

The criteria for crack growth which could lead to leakage of coolant into the plasma first assume that a surface crack exists. The crack is characterized by a hypothetical semicircular surface crack. The crack depth was taken to be the greater of 25% of the wall thickness or approximately an order of

TABLE 4.4-1

BLANKET MODULE FIRST WALL STRUCTURAL CRITERIA

Type of Failure	Mode	Criteria
Coolant Leakage Into Plasma	Crack Growth	$\Delta a \leq 0.10\ a_o$ Where a_o = Depth of a semicircular surface crack at BOL. Taken as 25% of the wall thickness or 0.025 cm, whichever is greater. Δa = Increase in crack depth from BOL to EOL
	Brittle Fracture	$K_{max} \leq 2/3\ (K_{IC})_{EOL}$ Where K_{max} = Maximum stress intensity factor from BOL to EOL $(K_{IC})_{EOL}$ = EOL Plane Strain Fracture Toughness
Excessive Deformation	Coolant Channel Gap Variation	$\Delta G \leq 0.30\ G_o$ Where G_o = Flow Channel Gap at BOL ΔG = Increase in Flow Channel Gap from BOL to EOL

magnitude larger (0.254 mm) than the mean grain diameter of 20% CW-316-SS assumed to be present at the beginning of life (BOL). As a result of cyclic loads the crack can slowly grow through the first wall to provide an opening for coolant leakage into the plasma before end of life (EOL). The coolant leakage criteria selected limit the crack growth to 10% of the BOL crack depth to provide a structural margin by limiting the total crack depth to less than one-half the wall thickness.

The criterion for brittle failure is assigned to prevent formation of an opening through the first wall. The criterion established prevents the postulated crack from initiating immediate first wall failure by brittle fracture due to the higher maximum stress intensity factor occurring at EOL. To prevent brittle fracture the maximum stress intensity factor is limited to two-thirds of EOL plane strain fracture toughness which is the lower limit of the fracture toughness of the material.

The criterion for excessive deformation is established to limit the EOL hot spot temperature to 493° C by limiting the coolant flow gap to 30% of the normal BOL flow channel gap. Limiting the hot spot temperature to 493° C will not result in accelerated crack growth or degraded material properties sufficient to cause accelerated crack growth or brittle fracture failure modes.

4.4.3 STRUCTURAL ANALYSIS

The structural analysis consisted of a determination of the various loads imposed on the module during normal operations. Once these loads were determined, a worst case duty cycle was established from which the stresses were determined. The loads considered in the analysis were those due to pressure, swelling, and thermal consideration and are summarized below.

PRESSURE LOADS

The blanket module first wall is subjected to an internal pressure caused by the helium coolant. Under normal operating conditions, the blanket module is considered to be pressurized to 5.52 MPa (54.4 atm) during plasma on-off cycling. Accordingly, the pressure load (P) is constant from BOL to EOL,

$$P = 5.52 \text{ MPa}$$

SWELLING LOADS

Under normal operation, the blanket module first wall is subjected to a fusion spectrum neutron flux during the plasma-on condition. For the specified wall loading of 4 MW/m2, the fusion spectrum develops a fast fusion neutron flux, $\phi = 1.2 \times 10^{15}$ n/s-cm^2, at neutron energies $E > 0.1$ MeV. With 1×10^5 cycles of plasma-on conditions at 19 minutes (1140 s) per cycle, the exposure time is 31,667 hours with an EOL fast fusion fluence ($E > 0.1$ MeV) of $\phi t = 13.25 \times 10^{22}$ n/cm^2. Because of the lack of fusion spectrum swelling data for 20% CW-316-SS the approach adopted was to assume that the fusion spectrum is no more severe than the fission spectrum with regard to swelling until fusion spectrum swelling data become available, or until uniform guidelines are established for the adjustment of the fission reactor swelling data. A 1.8% volumetric increase was found to occur at EOL at a first wall temperature of 450° C and was used in determining the combined stresses.

THERMAL LOADS

In normal operation, the blanket module first wall is subjected to a high surface particle heat flux during plasma-on conditions. For plasma-off conditions, the surface particle heat flux decays rapidly to zero. The normal operating surface heat flux is 1 MW/m^2.

Of interest in defining the thermal loads in the blanket module first wall are the steady-state and transient temperature distributions that occur during plasma on-off conditions. The procedure used to determine the temperature distributions consisted of deriving the global thermal response of the entire blanket module including the first wall, flow baffle, lithium, and lithium container. The thermal model does not, however, have the detailed nodal temperatures required for subsequent structural analysis. Accordingly, a detailed ANSYS thermal model of the first wall region alone was formulated which utilizes the local boundary conditions derived from the thermal calculation. The thermal solution was derived for the first wall region using the ANSYS program and the temperature distribution was saved on tape for subsequent recall in the structural analysis. The maximum through-the-wall temperature difference was found to be 73° C and occurred in the nose region.

WORST CASE DUTY CYCLE

Based on the loading analysis of the blanket module first wall under normal operating conditions, the worst case duty cycle consists of the following.

- Pressure — Constant at 5.52 MPa from BOL to EOL.
- Swelling — Variable and increasing to a volumetric increase of 1.8% at EOL.
- Thermal — Cyclic with a 73° C through-the-wall temperature difference.

4.4.3.1 STRUCTURAL CONSIDERATIONS

The structural analysis of the blanket module first wall was directed to deriving the BOL stresses and the effects of thermal and irradiation induced creep on BOL stresses during plasma on-off conditions.

BOL STRESS RESPONSE

The BOL stress response of the blanket module first wall to plasma-on pressure and thermal loading was derived using an ANSYS structural model with a pressure of 5.52 MPa and imposing the calculated thermal distributions. Similarly, the BOL stresses for plasma-off pressure and thermal loading were derived with a pressure of 5.52 MPa and a uniform temperature of 200° C.

The BOL stress distribution of the blanket module first wall for plasma on-off conditions indicated that the critical location is the hemispherical region adjacent to the nose. As the respective stress state is equibiaxial with a ratio of hoop to meridianal stress equal to unity, the hoop stress was selected to characterize the stress distributions during plasma on-off cycling. The hoop stress distribution through the wall is illustrated in Figure 4.4-1.

EFFECT OF THERMAL AND IRRADIATION INDUCED CREEP ON BOL RESPONSE

An elastic irradiation creep and swelling analysis was performed with a simple ANSYS model simulating the critical region in the hemispherical nose to assess the significance of swelling loads and redistribution of BOL stresses due to thermal and irradiation induced creep.

The analysis showed that the BOL stress response to pressure and thermal loads alone varied significantly from BOL to EOL response which included swelling loads and redistribution caused by thermal and irradiation induced creep. However, the maximum tensile stresses occur at BOL. The BOL and EOL stress response for plasma on-off conditions in terms of the hoop stress distribution through the wall at different times is illustrated in Figure 4.4-2.

With regard to deformation response from BOL to EOL, the simple ANSYS model shows a slow gradual outward growth to about 16,000 hours caused by thermal and irradiation induced creep relaxation of the pressure stresses. At 16,000 hours, the incubation period for 20% CW-316-SS is over and swelling begins. Thereafter, the first wall deformations increase more rapidly, reaching a maximum value of 0.046 cm at EOL.

4.4.4 STRUCTURAL EVALUATION

In order to perform a structural evaluation of the first wall in relation to failure resulting in coolant leakage a linear elastic fracture mechanics (LEFM) analysis is required. The analysis considers a hypothethical crack whose propagation is influenced by the stress intensity factors encountered in the material at BOL and EOL. The stress intensity factors are utilized in the appropriate equations to predict the rate of crack growth propagation to determine whether the crack growth is within the limits of the criteria previously specified. In addition, the maximum intensity factor which occurs at EOL can be compared with the linear elastic fracture toughness to provide a quantitative assessment of the susceptibility of the first wall to brittle fracture.

4.4.4.1 LINEAR ELASTIC FRACTURE MECHANICS

HYPOTHETICAL CRACK SIZE

A hypothetical semicircular surface crack is assumed to be present in the blanket module first wall adjacent to the hemispherical nose at BOL. Considering the BOL crack depth (a_0) to be 25% of the wall thickness (t), the crack depth is 0.04 cm. Since the minimum crack depth considered detectable by nondestructive testing (NDT) methods is 0.025 cm, the BOL crack depth (a_0) considered for the LEFM analysis is:

$$a_0 = 0.04 \text{ cm}$$

STRESS INTENSITY FACTOR SOLUTION

The stress intensity factor (K-solution) applied to shells of double curvature and considered to approximate the condition in the region of the hemispherical nose is given by

$$K = (M_K\sigma_t + M_b\sigma_b + 1.13\ P)\left(\frac{\sqrt{\pi a}}{\phi}\right)\left[f(\lambda)\right] \qquad (1)$$

Simplifying the K-solution for the geometry of the blanket module first wall and semicircular surface crack, when the module parameters and assumed crack size are considered, equation (1) reduces to:

$$K = 0.22\ (1.03\sigma_t + 0.70\sigma_b + 1.13P) \sim \text{MPa}\ \sqrt{\text{cm}} \qquad (2)$$

MAXIMUM AND MINIMUM STRESS INTENSITY FACTORS

The BOL response to pressure and thermal loads associated with plasma-on conditions showed inside and outside wall surfaces at the critical location to have stresses of +316.7 and -89.06 MPa as depicted on Figure 4.4-1. Accordingly, the maximum stress intensity factor (K_{max}) occurs at the inside surface where the linearized membrane (σ_t) and bending (σ_b) stresses for plasma-on conditions are:

$$\sigma_t = 113.81 \text{ MPa}$$

$$\sigma_b = 202.87 \text{ MPa}$$

Thus, utilizing the above stresses, the maximum stress intensity factor (K_{max}) at a pressure (P) of 5.52 MPa as determined from equation (2) is:

$$K_{max} = 59.65 \text{ MPa}\sqrt{\text{cm}}$$

Similarly, the BOL response to pressure and thermal loads corresponding to plasma-off conditions showed a relatively uniform +88.28 MPa for both inside and outside surfaces. Accordingly, the linearized membrane (σ_t) and bending (σ_b) stresses are:

$$\sigma_t = 88.28 \text{ MPa}$$

$$\sigma_b = 0 \text{ MPa}$$

Thus, the minimum stress intensity factor K_{min} at a pressure (P) of 5.52 MPa is:

$$K_{min} = 21.83 \text{ MPa}\sqrt{\text{cm}}$$

These values for K_{min} and K_{max} were used in the appropriate equations for determining fatigue crack growth and sensitivity to brittle fractures, discussed in the following paragraphs.

4.4.4.2 CRACK GROWTH

MATERIALS DATA

In order to provide an accurate structural evaluation of the blanket module first wall, fatigue and creep crack growth data simulating actual operating conditions are required but not available. Accordingly, only assumptions with regard to available materials data can be made at present.

The review of the available materials data(16) suggests that for the blanket module first wall constructed from 20% CW-316-SS operating in an inert environment and at a maximum metal temperature of 450° C with an EOL fast fusion fluence (E > 0.1 MeV) $\phi_t = 13.25 \times 10^{22}$ n/cm^2, the fatigue crack growth data in air at room temperature without correction for frequency/hold time, stress state, and irradiation should be used. Further, creep crack growth can be neglected. Over the low stress intensity factor range (Δ K), the fatigue crack growth rate $(\frac{da}{dN})$ can be expressed(17) in terms of the relation:

$$\frac{da}{dN} = 3.18 \times 10^{-21} (\Delta K)^{6.5318} \qquad (3)$$

CONTROLLED QUANTITY AND COMPARISON WITH CRITERION

To determine whether the first wall is protected against coolant leakage failure by the crack growth mode, the change in crack depth from BOL to EOL must be determined. The increase in crack depth (Δ a) is given by

$$\Delta a = \int_{a_0}^{a_f} \left(\frac{da}{dN}\right) dN \qquad (4)$$

In determining the fatigue crack growth rate $(\frac{da}{dN})$, a mean stress correction is applied to the stress intensity factor range (Δ K) in accordance with the NSM Handbook data for 316-SS in air at room temperature.

$$\Delta K = K_{max} (1 - R)^{0.35}$$

where

$$R = \frac{K_{min}}{K_{max}} = 0.37$$

Thus, utilizing the values of K_{min} and K_{max} previously determined,

$$\Delta K = 50.86 \text{ MPa} \sqrt{\text{cm}}$$

reducing equation (3) to:

$$\frac{da}{dN} = 4.45 \times 10^{-10} \text{ cm/cycle}$$

For small perturbations in the crack depth, the fatigue crack growth rate $(\frac{da}{dN})$ does not change significantly with the number of cycles (N) to require a piecewise integration. Accordingly, for the specified total of $N_T = 1 \times 10^5$ cycles, the increase in crack depth (Δ a) from equation (4) is given by:

$$\Delta a \cong \left(\frac{da}{dN}\right) N_T$$

$$\Delta a \cong 4.5 \times 10^{-5} \text{ cm}$$

The blanket module first wall criterion in protecting against coolant leakage in the crack growth mode is: $\Delta a \leq 0.1 a_o$. Since the initial crack depth (a_o) at BOL is 0.04 cm, the criterion is 0.004 cm. Accordingly, $\Delta a \ll 0.10 a_o$, and coolant leakage failure by the crack growth mode is not predicted for the blanket module first wall.

4.4.4.3 BRITTLE FRACTURE

MATERIALS DATA

The materials data required for the evaluation of the blanket module first wall in relation to brittle fracture are the EOL plane strain fracture toughness (K_{IC}). As such, the K_{IC} for 20% CW-316-SS in a helium environment at 450° C with an EOL fast fusion fluence ($E \geq 0.1$ MeV) $\phi_t = 13.25 \times 10^{22}$ n/cm^2 is required but not available.

The closest data base which exists for these conditions is for 20% CW-316-SS irradiated in the High Flux Isotope Reactor (HFIR). In 316-SS, HFIR irradiation actually overproduces helium relative to that which will occur in a CTR neutron radiation environment. Hence, if helium is assumed to be the source of the elevated temperature ductility loss in irradiated stainless steel, a consideration of these data can provide some estimate of the effects of irradiation on the plane strain fracture toughness.

Preliminary tensile data[18] are available for a neutron fluence of 2.38 x 10^{22} n/cm^2 ($E > 0.1$ MeV) at 450° C; this is equivalent to about 17 dpa or about 1.6 MW-years/m^2 first wall equivalent fluence. [This can be seen to fall considerably short of the target value of 13.25 x 10^{22} n/cm^2 ($E > 0.1$ MeV)]. In order to estimate K_{IC}, the HFIR tensile test data were used with the correlation:[17]

$$K_{IC} \approx \sqrt{2/3\; E\, \sigma_y\, \varepsilon_f\, (0.0005 + \varepsilon_u 2)}$$

where E = elastic (Young's) modulus
σ_y = tensile yield strength
ε_f = fracture strain (true)
ε_u = uniform elongation

For the tensile sheet specimens available from the HFIR irradiations, the total elongation was used in lieu of true fracture strain. Calculations of K_{IC} gave

$$K_{IC} = 384.7 \text{ MPa} \sqrt{\text{cm}}$$

For the purposes of the present blanket module analysis, this value was assumed to represent the EOL plane strain fracture toughness.

CONTROLLED QUANTITY AND COMPARISON WITH CRITERION

In protecting against coolant leakage failure by the brittle fracture mode, the blanket module first wall criterion is:

$$K_{max} \leq 2/3\ (K_{IC})_{EOL}$$

The maximum stress intensity factor (K_{max}) occurs at BOL during the plasma-on condition with a value of 59.65 MPa $\sqrt{cm}$. As the K_{IC} at EOL was 284.7 MPa $\sqrt{cm}$ the criterion is 256.5 MPa $\sqrt{cm}$. Accordingly, $K_{max} < 2/3\ (K_{IC})_{EOL}$, and coolant leakage by brittle fracture is not predicted in the blanket module first wall.

4.4.4.4 EXCESSIVE DEFORMATION

The structural evaluation of the blanket module first wall in relation to excessive deformation required an assessment of the EOL first wall deformation prior to comparison with the EOL flow channel gap change criterion.

FIRST WALL DEFORMATION

The simple ANSYS model fundamentally assumed a spherical shell loaded uniformly by internal pressure, through-the-wall thermal gradients, and swelling. A maximum temperature of 450° C was selected to simulate the local temperatures at the hemispherical nose. However, the swelling of 20% CW-316-SS is negligible at temperatures below 350° C. A review of the calculated first wall temperatures shows that only the local nose region is above the temperature at which swelling is expected to be significant. Accordingly, the swelling of the nose is restrained by the remainder of the hemispherical region. As such, it was found that actual swelling deformations will be approximately 40% of those predicted by the simple ANSYS model, i.e.,

$$\delta = 0.018 \text{ cm}$$

CONTROLLED QUANTITY AND COMPARISON WITH CRITERION

In protecting against excessive deformation failure by the change in flow channel gap (G) for this design, the blanket module first wall criterion is:

$$\Delta G \leq 0.3\ G_0$$

In the hemispherical region, the first pass flow channel varies from 0.13 cm at the cylinder junction to 0.076 cm at the nose. Accordingly, the nose flow channel region is minimum and establishes the 0.3 G_0 criterion limit as 0.023 cm. Now, the gap change (Δ G) is the first wall deformation (δ_E) of 0.018 cm at EOL. Since $\Delta G < 0.3$ G, excessive deformation failure by the flow channel gap change mode is not predicted in the blanket module first wall. The criterion may be altered or eliminated by future design changes should it be considered necessary to reduce the sensitivity of the design to coolant gap configuration.

4.4.4.5 SUMMARY

The blanket module first wall structural evaluation, based on the current HFIR and LMFBR data base for 20% CW-316-SS and an EOL fast fusion spectra fluence (E > 0.1 MeV) $\phi_t = 13.25 \times 10^{22}$ n/cm^2, shows the current design to be acceptable. A summary of the blanket module structural evaluation is presented in Table 4.4-2.

TABLE 4.4-2

BLANKET MODULE STRUCTURAL EVALUATION SUMMARY
NORMAL CONDITIONS

Failure Mode and Structural Criteria		Allowable Value	Calculated Value	MS*
Coolant Leakage Into Plasma	Crack Growth (cm)	0.004	4.5×10^{-5}	88
	Brittle Fracture (MPa $\sqrt{\text{cm}}$)	256.5	59.65	3.30
Excessive Deformation of Coolant Channel (cm)		0.023	0.018	0.28

$$\text{*MS} = \frac{\text{Allowable Value}}{\text{Calculated Value}} - 1$$

4.5 MAGNETIC LOAD ANALYSIS

4.5.1 PURPOSE AND SCOPE

The purpose of the analysis performed for this study was to develop insight as to the magnitude of the loads to which the modules could be subjected in the presence of the magnetic fields required during the reactor operation. Eddy currents of significant magnitude could be induced in a lithium blanket when the poloidal field is pulsed. This system of induced currents would in turn interact with the ambient confining fields to produce a system of magnetically induced body forces on the modules. The magnitude of these forces as a function of module size was investigated to determine their potential impact on the blanket design. The results of this study are discussed.

4.5.2 ANALYTICAL RESULTS

The analysis which was performed is documented in Reference 19. In the analysis, a cylindrical container filled with the lithium conductor was examined to determine the characteristic response in terms of magnetic fields, resistivity of the conducting lithium, and characteristic dimensions of the modules. The characteristic dimension is defined as the radius of a cylindrical module. Typical results are given in Table 4.5-1, which considers module orientations which provide the highest values of mechanical loads. Note that the forces per unit length of module increase by the third power of the characteristic dimension. For torque reactions, which tend to twist the module about its principal axis or which tend to bow the module along its principal axis, the torque per unit length will vary as the fourth power of the characteristic dimension. The container stresses would vary by a lower power since the larger modules would by virtue of their larger cross section have more available material to withstand the loads. Note that a module support structure loaded with a few large modules will be subjected to higher net structural loads than one loaded with a larger number of smaller modules.

Since magnetically induced loads were found to vary as the third or fourth power of the characteristic dimension, judgment must be exercised in selecting the size of the blanket module to be certain that the structure is adequate for withstanding the load imposed by the module.

TABLE 4.5-1

MECHANICAL RESPONSES FOR CONTAINERS WITH LIQUID LITHIUM*

CONTAINER RESPONSE		UNITS	S = RADIUS OF CYLINDER IN METERS 1.0	0.2	0.05
MAXIMUM LOCAL PRESSURE	$1/2\ \alpha\ S^2$	Pascal	320,000	12,800	800
This occurs locally either at the center of the end of the container or on one of the container side walls. The occurrence is affected by the sign of B depending on whether poloidal flux is increasing or decreasing.					
MAXIMUM FORCE PER METER	$\frac{\alpha}{2}\ S^3$	Newtons/m	320,000	2,560	40
This is the maximum tensile load in a circumferential direction.					
MAXIMUM OVERTURNING MOMENT PER METER OF LENGTH	$\pi/8\ AS^4$	Newton-m/m	2.08×10^6	3,330	13
This is the moment tending to bow the container along its length and is a response to poloidal fluxes pulsing along the container axis.					
MAXIMUM TWIST PER METER	$\pi/4\ AS^4$	Newton-m/m	4.19×10^6	6,700	26
This is the response assuming both poloidal pulsing flux and toroidal constant flux normal to the container. The moment tends to rotate the container about its longitudinal axis.					

*For this example:

$B_\perp$ = Toroidal Field $\quad$ B = Poloidal Field

$B_\perp = 10$ T $\quad$ B = 1.2 T $\quad$ $\dot{B} = 0.24$ T/s

$A = \frac{B_\perp \dot{B}}{\rho}$ $\quad$ $\alpha = \frac{B\dot{B}}{\rho}$ $\quad$ S = Characteristic Dimension $\quad$ ρ = Resistivity

$\rho = 45 \times 10^{-8}\ \Omega$-m $\quad$ $\alpha = 640{,}000$ $\quad$ $A = 5.3 \times 10^6$

The relative merits of a small module (the reference cylindrical module with characteristic dimension = 0.05 m) with respect to induced load are readily apparent from Table 4.5-1.

4.6 TRITIUM BREEDING

Detailed neutronics calculations to determine tritium breeding ratio were deliberately excluded from the scope of the FY 78 effort. However, one-dimensional transport calculations were performed to estimate the tritium breeding ratio for the reference blanket module concept. The calculations indicated a breeding ratio of slightly greater than 1.1. Since the goal was a breeding ratio of 1.2, the module length of the reference concept would need to be lengthened or otherwise modified to achieve the desired breeding ratio. Table 3.3-1 shows that for the lithium blanket module design concepts which were reviewed, the breeding ratios vary from > 1 to 1.3, where the blanket modules ranged in thickness from 0.5 m to 0.8 m. It appears reasonable, therefore, that the 1.2 breeding ratio can be achieved in the cylindrical module design by lengthening the module to increase the lithium volume or incorporating some other design modifications.

5.0 DESIGN ASSESSMENT

This section provides an assessment of performance of the reference blanket concept for conditions beyond the guidelines previously defined for nominal operation. Additional thermal analyses were performed for off-design conditions, for changes in operating cycle time, and for a case where incorporation of a divertor is considered. A qualitative lifetime assessment was also made based on a comparison of the detailed structural evaluation performed in support of the concept development. The structural evaluation is presented in more detail in Reference 16. In addition, a limited assessment was made relative to routine service and maintenance. Results of the assessment are presented in the following sections and summarized in Table 5.1-1.

5.1 ASSESSMENT OF OFF-DESIGN PERFORMANCE

Thermal and hydraulic performance and structural assessment of the design with respect to lifetime design conditions were presented in Section 4.0. These results showed that the cylindrical module design had a positive margin of safety for a neutron wall loading of 4 MW/m^2 and an associated heat flux loading of 1 MW/m^2. The maximum first wall temperature was maintained at 450° C with an associated pumping power of 2.2%. Off-design conditions could be expected to perturb either the temperature or pumping power. Thermal analyses were performed for the off-design conditions associated with postulated changes in coolant flow, reduced power operation, and hypothetical plasma disruption as described in the following section. Results are discussed in the following sections.

5.1.1 VARIATIONS IN COOLANT FLOW

5.1.1.1 EFFECT ON THERMAL PERFORMANCE

Steady-state thermal conditions of the module at various helium flow rates

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TABLE 5.1-1

SUMMARY OF MODULE RESPONSE FOR VARIOUS OPERATING CONDITIONS

Condition	Helium Coolant Outlet Temp., °C	First Wall Peak Temp., °C	Max. Li Temp. °C	First Wall Margin of Safety			Pumping Power
				Crack Growth	Brittle Fracture	Excessive Deformation	
Normal Design	435	452	627	88	3.3	0.28	2.2
10% Flow Reduction	465	470	655	∿ 88	∿ 3.3	∿ 0.28	2.0
Loss of Coolant to 1% in 1 s	448 @ 1 s[(a)]	690 @ 1 s[(a)]	627 @ 1 s[(a)]	∿ 88[(b)]	∿ 3.3	∿ 0.28	N/A
Reduced Power (50% of Normal @ 34 atm Coolant Pressure)	450	420	550	8826	14.7	2.4	1.1% @ 34 atm coolant 0.5% @ 54.5 atm coolant
Hypothetical Plasma Disruption[(c)] (0.010 s Pulse, 5 MW/m^2, 1×10^4 cycles)	439	500	627	43	2.5	∿ 0.28	N/A
Change in Cycle Duration							
190 minute Pulse, 10^4 Cycles	435	450	630	880	3.3	0.28	∿ 2.2
2 minute Pulse, 10^6 Cycles	< 435	< 450	< 630	8.8	3.3	0.28	∿ 2.2
Incorporation of a Divertor	440	270	633	∞	11.0	∞	1.25

(a) Function of time to shutdown per Figure 5.1-2.

(b) Margins of Safety for 400 ms Shutdown.

(c) Function of Heat Flux Assumed per Figure 5.1-3, Values shown for 5 MW/m^2.

were calculated. The steady-state first wall, inner cylinder helium exit, and lithium peak temperatures for coolant flows ranging from 20% to 140% of the nominal design value are presented in Figure 5.1-1. The effect of a 10% variation in nominal flow shows that the critical first wall temperature is increased by 20° C when coolant flow is reduced or decreased by 10° C when the coolant flow is increased. The effect on lifetime is discussed in the following section. If the flow rate should be significantly reduced beyond 10%, the first wall temperatures would increase to levels above those considered to be acceptable in this study.

A case was postulated for loss of coolant in which the helium coolant flow is suddenly reduced to 1% of full flow in one second. The first wall temperature in this case would lead to a temperature rise of the inner and outer cylinder as shown in Figure 5.1-2. The large temperature rise indicates that a quick sensing control system, a redundant coolant system, or other compensatory design features will be required in the system design to preclude reaching such unacceptable temperatures, or measures must be taken to preclude the possibility of such a postulated accident.

5.1.1.2 STRUCTURAL EFFECT

The 20° C temperature increase of the first wall (to a maximum temperature of 470° C) due to a 10% reduction in coolant flow is not considered significant in accelerating crack growth or brittle fracture failure modes as discussed in the structural evaluation under normal conditions in Section 4.4.5. This is true since the temperature of 485° C due to changes in the flow channel gaps caused by irradiation creep and swelling of the first wall is considered acceptable. It is expected that the margins of safety previously stated in Section 4.4.5 for crack growth and brittle fracture would be reduced, but not significantly. However, fatigure crack growth and plane strain toughness data for 20% CW-316-SS in the 450° C to 500° C temperature range are required to accurately assess the importance of increased first wall temperatures or potential coolant leakage.

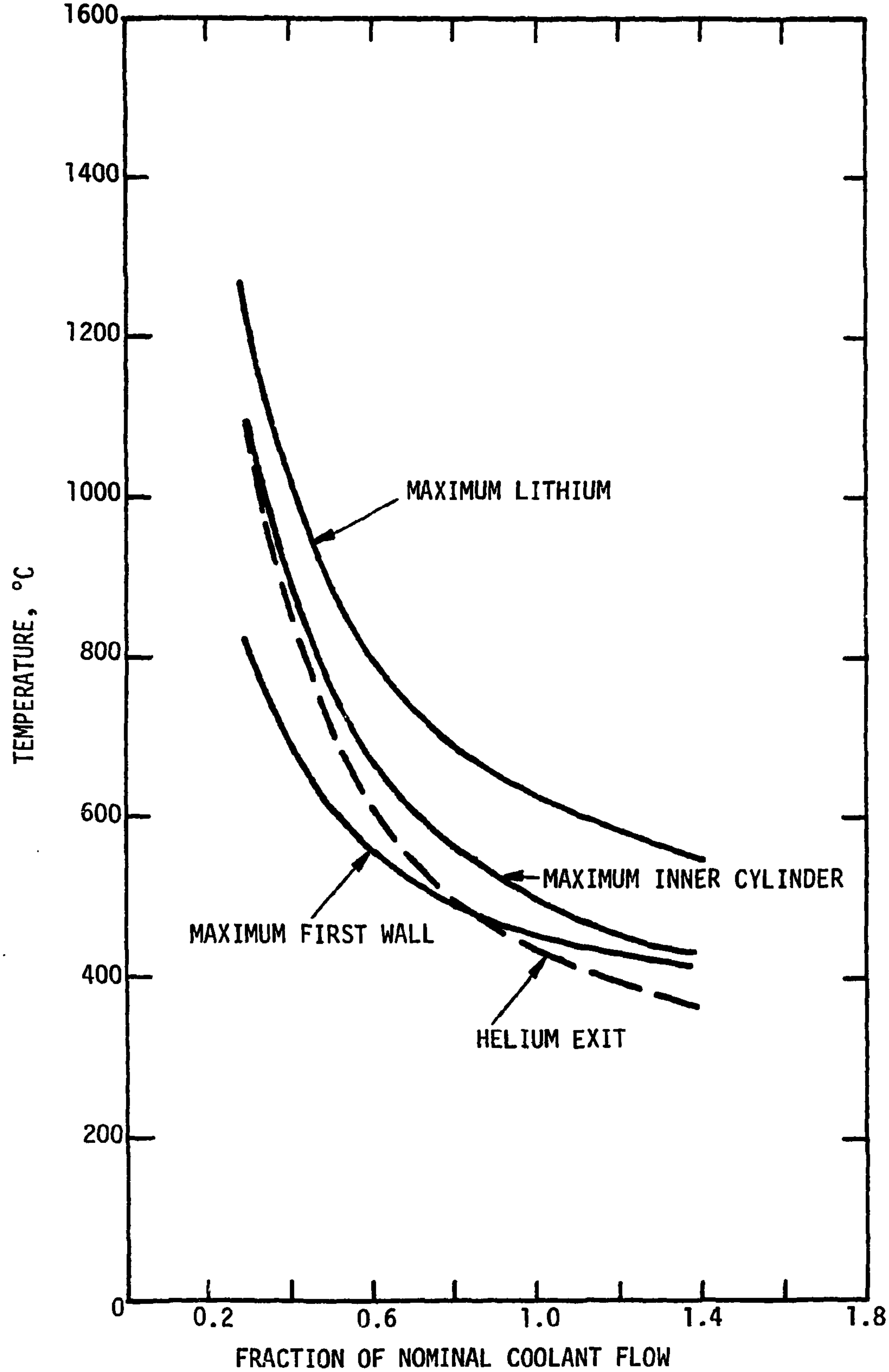

Figure 5.1-1. Effect of Variation in Coolant Flow on Module Temperature

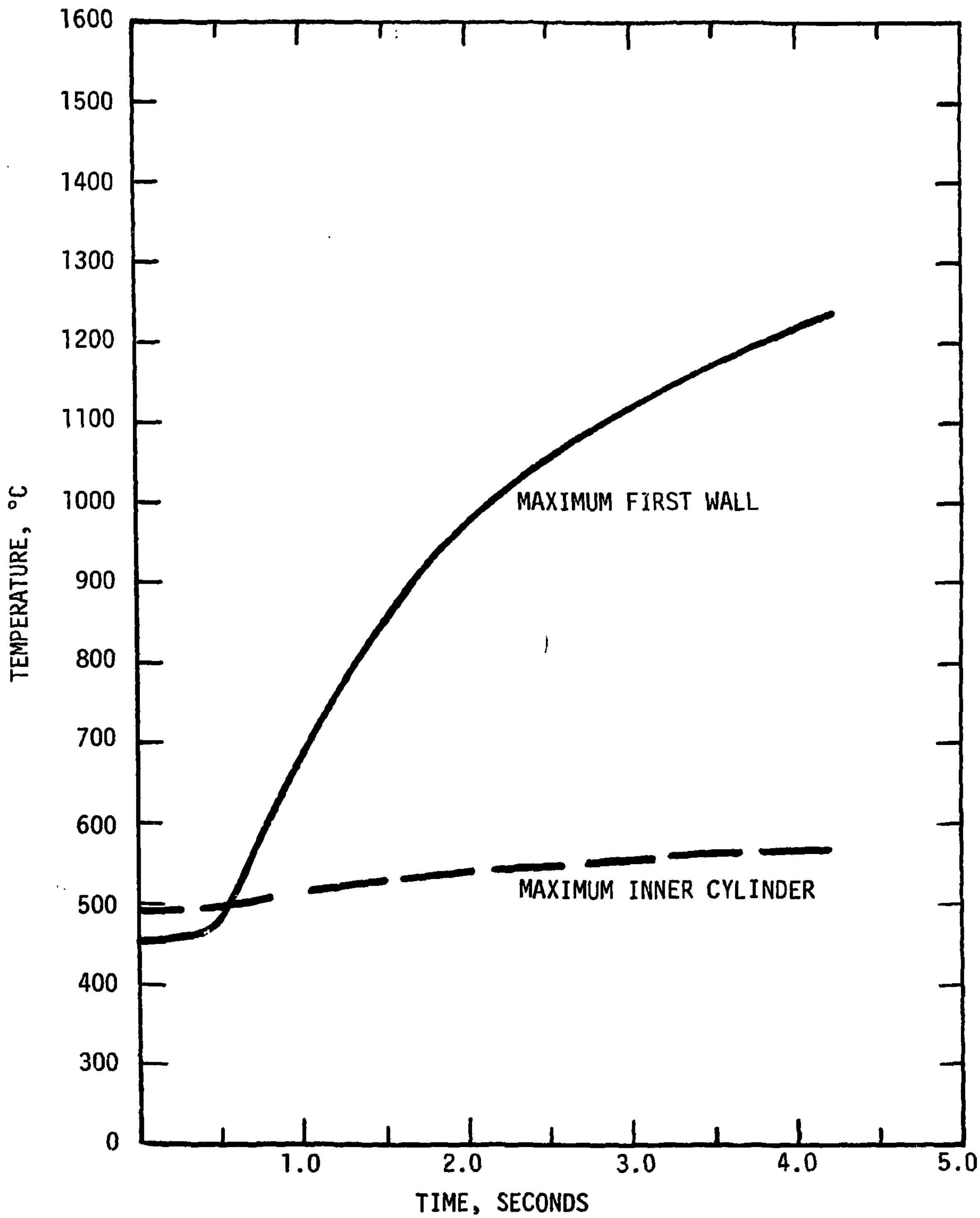

Figure 5.1-2. Structural Temperature Responses to Loss of Coolant to 1% of Nominal Flow in One Second

5.1.2 PART POWER CONDITIONS

5.1.2.1 EFFECT ON THERMAL PERFORMANCE

A part power condition of 2 MW/m^2 wall loading (0.5 MW/m^2 associated particle heat flux) was investigated to determine the effects of postulating lower power levels. At this condition, the inner cylinder would reach a design temperature limit of 500° C when the first wall reaches 420° C. For a first wall temperature of 450° C, the inner cylinder temperature would be 575° C. For this case, cooling can be achieved with lower pressure helium at 34 atm, with corresponding inlet and outlet temperatures of 200° C and 450° C. The pumping power requirement is reduced to 1.1%. If the coolant pressure is 54.5 atm as used in the reference design, the required pumping power would be further reduced to 0.5%.

5.1.2.2 STRUCTURAL EVALUATION

Since the reference design is essentially fluence limited with respect to lifetime assessment, the lower neutron flux level and lower pressure and thermal stresses would lead to a significant increase in first wall structural life, particularly with respect to cracks where the margin of safety increases approximately a hundred-fold. The lithium inner cylinder may now be the component which governs the lifetime of the module and will require analysis to determine the lifetime or margins of safety.

5.1.3 HYPOTHETICAL PLASMA DISRUPTION

A parametric analysis was performed to determine the first wall temperature response if the module is subjected to hypothetical particle heat fluxes substantially higher than 1 MW/m^2 for 0.010 s. The maximum first wall surface temperature and the maximum average wall temperature for the range of surface heat fluxes (up to 80 MW/m^2) studied are shown in Figure 5.1-3. It is seen that the first wall surface temperature has not reached the melting point ($\sim$ 1400° C) with a heat flux of about 80 MW/m^2 for 0.010 s, but the maximum average wall temperature is still very low. Analysis with a more detailed model is required to determine the temperature distributions through the wall.

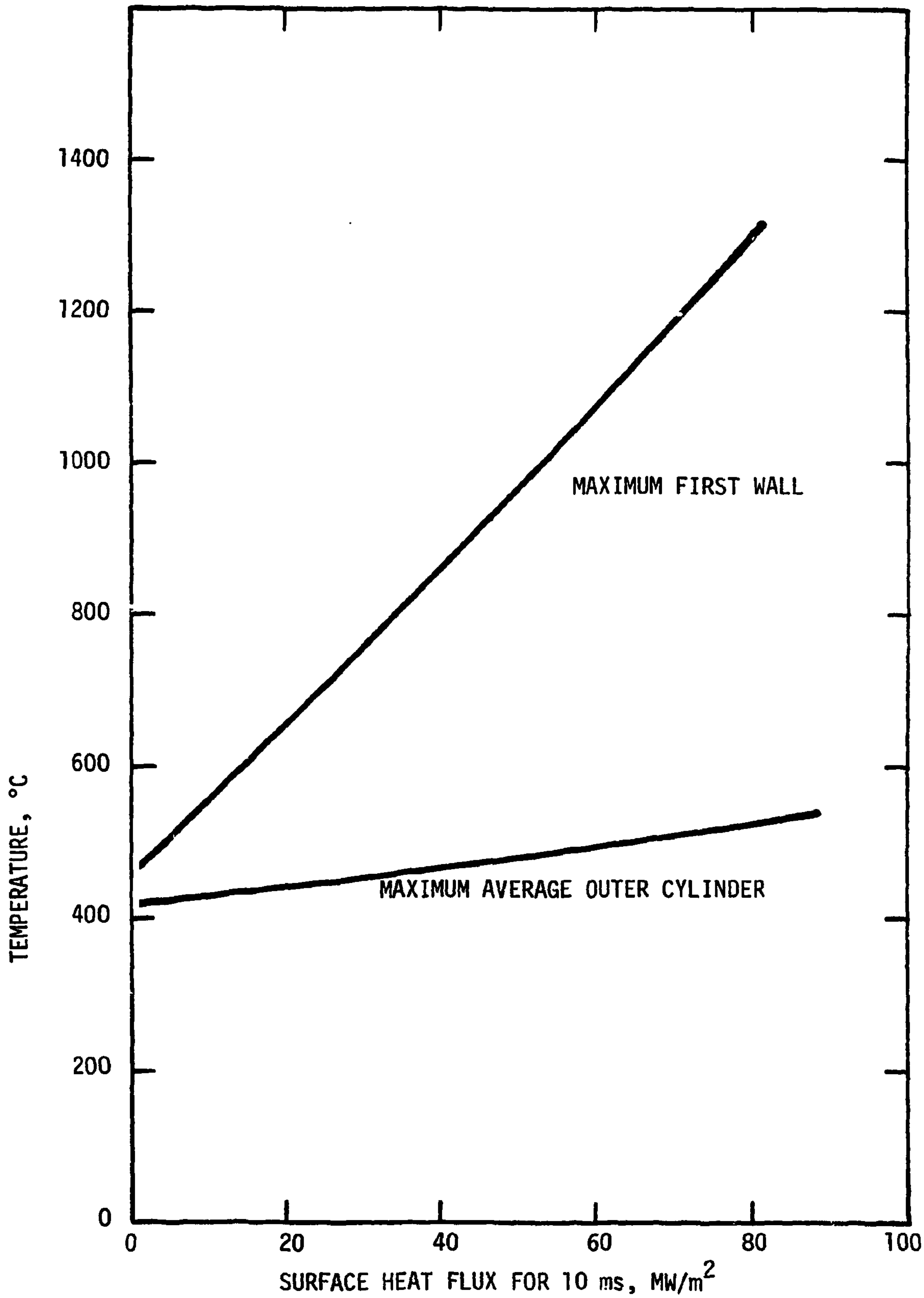

Figure 5.1-3. First Wall Temperature Response to Heat Flux

The effect of such a thermal excursion on the reliability of the blanket design needs to be assessed. In particular, the loss of cold worked material properties through the wall needs to be determined. Significant additional study is obviously necessary to determine in a realistic way the effects of postulated disruptions and then consider how acceptable disruptions might be accommodated within a blanket design. Structural lifetime margins of safety for a 5 MW/m^2 heat flux were calculated and are presented in Table 5.1-1.

5.2 ASSESSMENT OF CHANGE IN CYCLE DURATION

5.2.1 EFFECT ON THERMAL PERFORMANCE

The reference pulse cycle consisted of 19 minutes plasma-on and 1 minute plasma-off for a 20 minute cycle at a 95% duty. The effect of the cycle duration variation on the module thermal performance was investigated by considering the following two extreme cycle times: 2 minutes of burn-time with a 1 minute off-time and a 190 minute burn-time with the same 1 minute off-time.

As discussed in Section 4.3.4.4 on transient characteristics, the thermal conditions of the module require about three minutes to essentially reach the reference design steady-state conditions after a plasma-off period of 1 minute. If the burn-time is reduced to 2 minutes, the helium exit temperature would not reach the peak design value at the end of the 2 minute pulse. This results in a continuously changing coolant temperature at the steam generator inlet. After many cycles, the helium exit temperature increase and decrease would stabilize to lower levels than in a cycle with a longer burn time. The net effect is to decrease the performance of the power conversion system.

If the burn time is 190 minutes (compared to 19 minutes) with the same 1 minute off-time, there would not be any effect on the thermal performance of the module. The module would approach the steady-state condition in a few minutes after the 1 minute plasma-off period (see Figure 4.3-5) and the temperatures would be maintained throughout the remainder of the 190 minute burn. The overall efficiency of the power cycle would be slightly improved since there is a lower percentage of "off-time" and the net average helium temperature to the power conversion system would be higher than for the 19 minute burn.

5.2.2 STRUCTURAL EVALUATION

Potential coolant leakage of the module was shown to be by the brittle fracture mode of failure, considering EOL plane strain fracture toughness of the irradiated material, which is in turn dependent on the EOL fast fusion fluence. Therefore, the total time associated with plasma-on conditions provides a basis for assessing brittle fracture. The produce of the plasma-on time (t_{on}) and the number (N) of on-off cycles is a constant, i.e.,

$$t_{on} N = C$$

Based on the current High Flux Isotope Reactor (HFIR) data [16] and the extensions adopted in the analysis of Section 4.4 for the first wall material, the normal condition based on a plasma duration of 19 minutes and 10^5 cycles gives:

$$t_{on} N = 19 \times 10^5 \text{ minute-cycles}$$

Consequently, an extended cycle time would dictate a lower number of cycles to maintain the same fluence and margin of safety in the brittle fracture mode. Conversely, a larger number of cycles would be permitted if the cycle time were reduced. Extending the cycle time from a plasma-on duration of 19 minutes to 190 minutes reduces the allowable number of cycles from 1×10^5 to 1×10^4, although the lifetime at power is still the same.

Similarly, shortening the plasma-on duration from 19 minutes to 2.0 minutes increases the allowable number of cycles from 1×10^5 to $\sim 1 \times 10^6$.

The margin of safety for brittle fracture remains unchanged at the value of 3.3 presented in Table 4.4-2 for both the extended and shortened cycles, whereas the margin of safety relative to crack growth will be increased or reduced by a factor of 10, respectively, compared to the MS = 88 for the 19 minute plasma-on cycle.

5.3 ASSESSMENT OF INCORPORATION OF A DIVERTOR

5.3.1 THERMAL PERFORMANCE

If a divertor were incorporated, the first wall would be subjected to some (substantially reduced) particle heat flux. However, for the purpose of this study the heat flux was assumed to be zero, thereby permitting a comparison between the reference case with no divertor and a case where the divertor is considered 100% efficient. Steady-state thermal conditions for the module were calculated, therefore, with the total energy deposited in the blanket module coming solely from the 4 MW/m^2 neutron wall loading. The thermal performance for this case is presented in Figure 5.3-1. In the absence of the surface heat flux, the module operation is limited by the 500° C maximum inner cylinder temperature, as shown in the figure. At this condition, the helium coolant exit temperature is 440° C, comparable to the reference case, but the maximum front wall temperature is only 270° C as compared to 450° C in the reference case.

However, since a high coolant flow is no longer required to cool the first wall, the required pumping power is reduced to 1.25%. These operating temperatures and pumping requirements are readily apparent from the points (circles) identified on the curves. The effect of a totally efficient divertor, therefore, is to reduce the pumping power to a value of 1.25%. If desired, a trade-off could be effected between pumping power and structural lifetime, since the module can operate at lower coolant pressure load (and consequently a lower stress) if the 2.2% pumping power of the reference case is accepted.

5.3.2 STRUCTURAL EVALUATION

Based on the relatively low first wall temperature of 270° C, the margins of

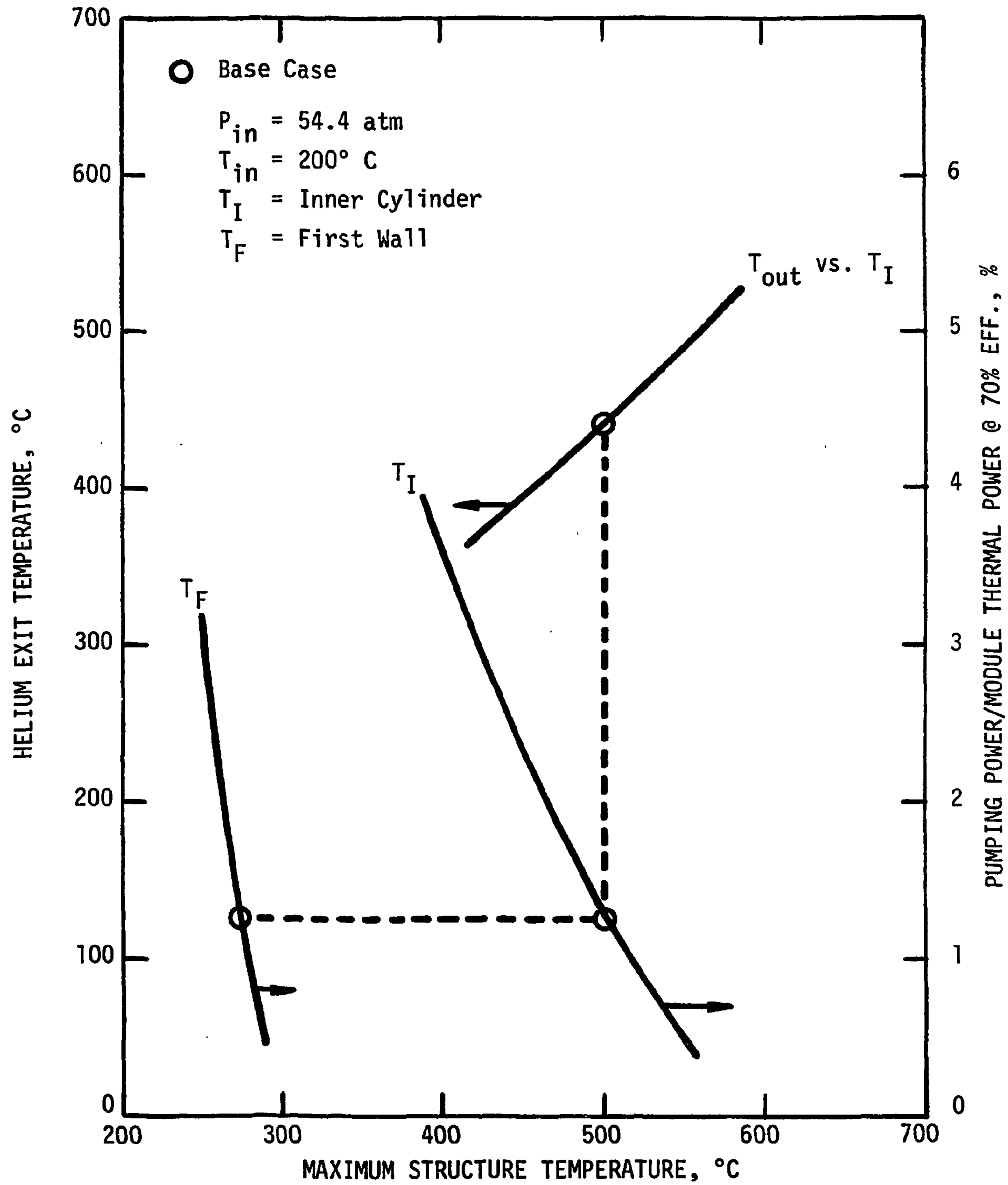

Figure 5.3-1. Thermal Operating Limits of Module with 100% Efficient Divertor

safety in protecting against coolant leakage will increase over the reference case (without a divertor). The margin of safety in protecting against crack growth is very large. Fatigue crack growth is not expected to occur, since the cyclic thermal stresses across the first wall approach zero, and creep crack growth is negligible at a 270° C first wall temperature. An increase in margin of safety for brittle fracture is realized since the plane strain fracture toughness at 270° C is higher than at 450° C. The increase is achieved although the blanket module is still pressurized to 5.5 MPa (54.4 atm) and the first wall irradiated fast fluence (E > 0.1 MeV), $\phi_t = 13.25 \times 10^{22}$ n/cm^2, at EOL remains unchanged because the fracture toughness is expected to be higher at the 270° C first wall temperature. The potential hot spot problem due to radiation induced creep and radiation induced swelling, discussed in Section 4.4, is reduced since creep will be small and swelling will be essentially nonexistent at the lower first wall temperature. Incorporation of a divertor does increase the margins of safety of the first wall above those identified for the design without a divertor.

With the higher margins of safety for the first wall the lithium inner cylinder, as noted for the part power condition (Section 5.1.2), may now be the component which governs lifetime of the module and will require analysis to determine the actual life.

5.4 ASSESSMENT OF ROUTINE SERVICE MAINTENANCE

5.4.1 DESCRIPTION OF ASSEMBLY/DISASSEMBLY PHILOSOPHY

The service and maintenance philosophy behind the cylindrical blanket module concept is based on a modular rather than a unit assembly/disassembly operation. In practice the blanket assembly of the reactor is made up of 48 D-shaped structural segments, six of which are shown in Figure 5.4-1. To service the cylindrical blanket modules attached to the D-shaped segment, the segment is first removed from the reactor.

The structural D-shaped segment containing the defective blanket module is identified and the supply service lines are disconnected. The lithium lines

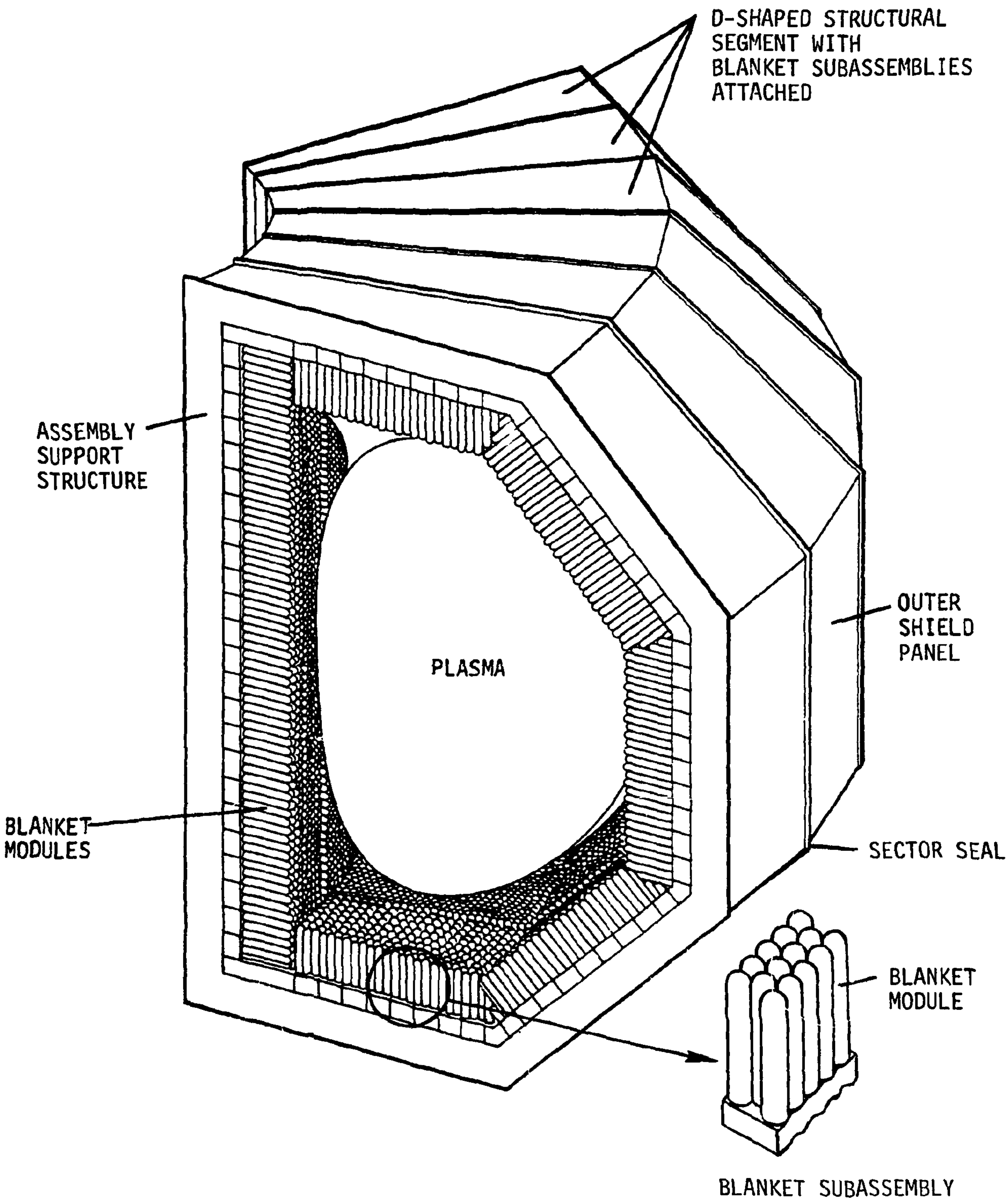

Figure 5.4-1. Blanket Sector Assembly

are capped to prevent loss of the liquid lithium from the blanket modules. The seal welds or bolted flange arrangements used to unitize the 48 segments are decoupled, freeing the segment or segments to be serviced from the remainder of the blanket assembly. The segment located midway between two adjacent TF coils is then removed radially outward from the reactor and transported to a preplanned work area. If the defect is in a D-shaped structural segment adjacent to the segment located midway between TF coils, the disassembly process is continued by removing the second segment. Since these segments are partially encompassed by the TF coils, they must first be translated sideways to the midposition and then moved radially out from the reactor. The defective segment is replaced under the TF coils by a reversal of the removal procedure and the center segment is replaced. The reactor is made operative by replacing the segments and connecting the necessary supply lines and rebolting or welding the seals of the segments.

In a parallel effort, the D-shaped structural segment containing the blanket module to be serviced is further disassembled. The outermost shield panel of the D-shaped structural segment in line with the defective blanket subassembly is removed by cutting the necessary seal welds. Removal of this shield panel makes possible access to the module subassembly manifolds and mounting bolts. Connections from the blanket subassembly (Figure 5.4-2) and the D-shaped structural segment header pipes are disconnected. After the module subassembly is attached to a handling fixture, the bolts securing the module are removed. By means of manipulating the blanket subassembly handling fixture, the subassembly is removed. Removal of more than one subassembly may required if the assembly to be serviced is located at sections of the D-shaped structural segment where orientation of the subassembly changes. The required subassembly is replaced with a new unit. The refurbished D-shaped structural segment is then tested and placed in storage as a replacement unit. At this time the condition of the questionable subassembly is further evaluated and a decision made whether further repairs are warranted or the entire assembly is to be discarded.

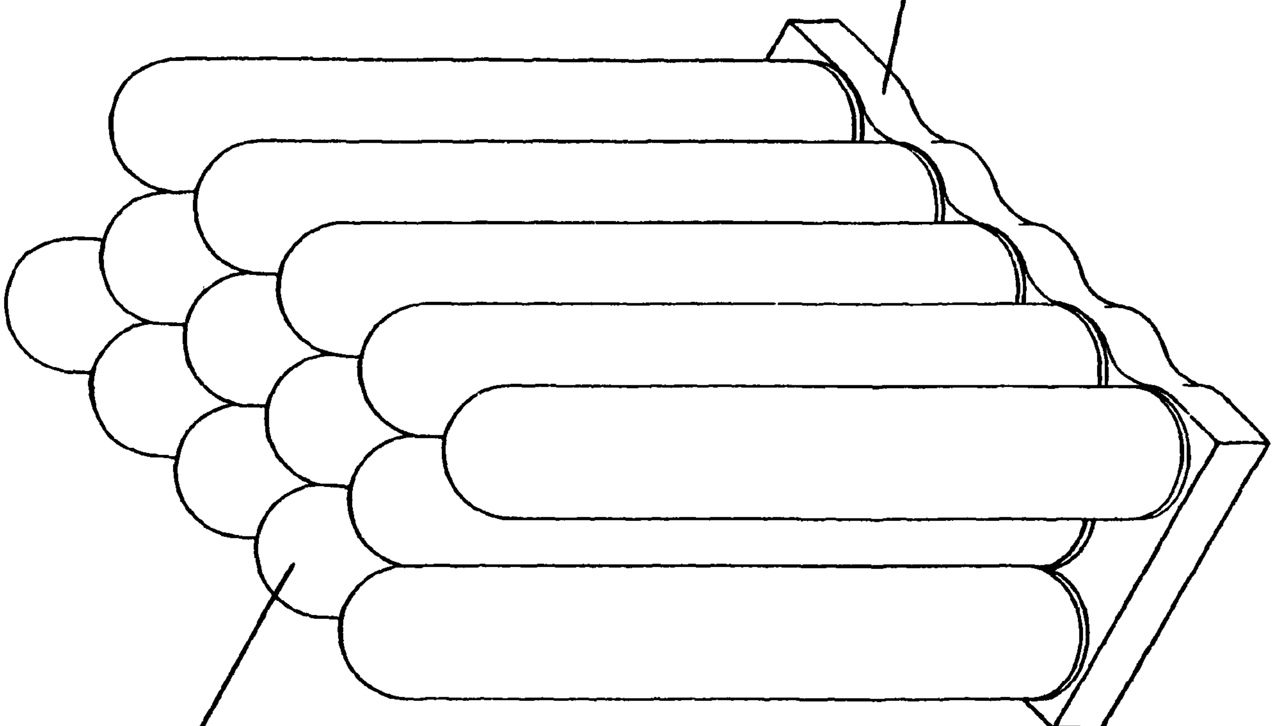

Figure 5.4-2. Typical Blanket Subassembly

All disassembly and assembly operations described in the above paragraphs will be performed by remote handling equipment. With this requirement as one of the design parameters, the cylindrical blanket module was designed so that it could be disassembled by starting at the outer periphery of the structural segment and working radially inward. Circular seal welds are used wherever possible so that existing technology on remote circular cutting and welding machines could be utilized. Manifolds are designed in two halves. The outer half is removed to expose the circular welds joining the feeder lines of the individual modules to the bottom of the lower half of the manifold.

5.4.2 ASSEMBLY AND DISASSEMBLY EQUIPMENT REQUIREMENT CONSIDERATIONS

Servicing and maintaining the cylindrical blanket module concept as described in earlier chapters will require hands-off operation. This requires shielded vehicles containing remote manipulators to operate in the vicinity of the reactor. Cranes and track wheeled dollies capable of supporting a minimum of 50 tonnes will also be required in the reactor compartment.

Work and storage areas for the D-shaped structural segments will be required near the reactor compartment. These areas must be shielded and will be serviced by 50 tonne cranes and contain manipulators for performing assembly/disassembly operations on the D-shaped structural segments and the blanket subassemblies. Remotely controlled circular welding and cutting machines capable of cutting and welding various diameter seal welds will be required in the work area. Cutting and welding machines capable of making longitudinal cuts and welds will also be required. Hydraulic pressure testing facilities with capacities of approximately 68 atm (1000 psi) for testing the reassembled blanket modules and associated manifolds should also be available in the work area. A piping and storage system for charging and discharging the blanket module lithium cylinders will also be required.

6.0 CONCLUSIONS AND RECOMMENDATIONS

Based on the analysis performed in support of the concept selected in this study, a viable blanket concept was developed which warrants further development and design refinement. This cylindrical module concept meets the goals of the study to produce a blanket concept which operates under a reasonable set of reactor conditions and advances the state of the art of blanket concepts by considering reliability, thermal performance, structural lifetime, helium generation, and tritium breeding. To the extent permitted, consistent with the scope of the study, the design was shown to perform satisfactorily against the selection criteria and the pertinent design and performance goals developed during the study. The following is a list of significant conclusions resulting from the study.

- Stainless steel is a viable structural material for neutron wall loading and first wall heat flux of 4 MW/m^2 and 1 MW/m^2 respectively. Under these conditions, the design meets the goal of 10^5 cycles of 20 minutes with 95% duty, based on considerations of crack growth and brittle fracture at 450° C operating temperatures.

- This concept can reliably withstand full coolant pressure; thus, it satisfactorily addresses the key requirement of preventing breach of the lithium container in the event of failure of high pressure coolant circuits, which was a concern in the earlier designs.

- This method of first wall cooling is not sensitive to the accumulation of helium generated within the lithium which compromised cooling in earlier designs.

- The concept is structurally efficient, amenable to analysis, and simple in shape; can be readily fabricated and evaluated by testing; and is adaptable to mass production.

- Structural support of the modules as replaceable subassemblies is judged a reasonable approach to assembly and maintenance and is compatible with remote handling techniques.

- First wall temperatures of $\sim$ 450° C can be achieved with 200° C inlet temperature at $\sim$ 2% pumping power with reasonable helium exit temperatures ($\sim$ 450° C) compatible with acceptable power conversion.

- The main support structure is designed to operate at helium inlet conditions and thereby minimizes thermal growth of the structure and relative motions between the blanket assembly and interfacing piping.

- Sealing between the plasma and the outer vacuum boundary can be achieved.

- Scoping analysis indicates that a tritium breeding ratio of $\sim$ 1.2 can be obtained.

It is recommended that the design be further developed, particularly in the areas of structural support, maintenance, and piping systems. The following are recommendations to support further work in the areas of design and development, analysis, and testing.

DESIGN AND DEVELOPMENT

- Improve the module design and packaging to improve thermal and neutronic performance, and reduce the number of modules, penetrations, manifolds, connections, and feet of piping to enhance the system reliability, consistent with meeting structural lifetime requirements.

- Identify the blanket system component failure modes and perform a reliability assessment of the design with regard to its performance and operation in utility service.

- Fabricate a module to verify that the 20% CW material can be produced in the design configuration and to verify flaw detection capability to support the analytical assumptions for flaw growth propagation.

- Produce models and mock-ups to verify assembly/disassembly capability and develop a better perspective relative to piping design and interfacing components.

ANALYSIS

- Perform more detailed analysis to better quantify MHD interactions to provide definition of realistic structural responses of the blanket due to pulsing magnetic fields.

- Perform detailed neutronics analysis to verify breeding capability.

- Analyze the design to determine compatibility with high vacuum techniques.

- Calculate the effect of reflectivity of higher modes of cyclotron radiation to determine if there is any adverse effect on the plasma temperature because of the stacked module arrangement.

TESTING

- Simulate the geometry of the flow paths in the region of the nose of the module and perform tests to verify the pressure loss and the heat transfer coefficients to support the thermal analysis assumptions relative to first wall cooling and pumping power.

- Perform fatigue and creep crack growth testing of simple representative stainless steel surface crack specimens in the 400-500° C temperature range to verify structural analysis assumptions.

- Define and implement a full-scale module test with the appropriate thermal heat input to the first wall to verify module performance.

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