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DEMONSTRATION TOKAMAK HYBRID REACTOR (DTHR) BLANKET DESIGN STUDY

DECEMBER, 1978

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fusion power systems department

Westinghouse Electric Corporation

P.O. Box 10864, Pgh. Pa. 15236

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ABSTRACT

The Demonstration Tokamak Hybrid Reactor (DTHR) was conceived as a near-term fusion engineering test reactor. Its principal missions are the demonstration of fissile fuel production in a tokamak hybrid reactor, the integration of the critical technologies involved and the provision of a flexible engineering test bed for reactor materials and components. This report presents the evolution and selection of a potentially attractive DTHR fertile blanket concept, which makes significant use of proven light water fission reactor technology. The concept has the potential for producing large amounts of fissile fuel and electric power, with tritium breeding in either separate or upgraded blanekts. Tritium breeding was considered to be feasible in the inner blanket or outboard of the fissile breeding zone. The blanket incorporates thorium oxide fuel, clad with zircaloy and cooled by radial inflow of boiling water in a rectangular pressure vessel operating at moderate pressures. This flow configuration satisfies the thermal-hydraulic, mechanical and structural design goals and produces fissile breeding performance comparable to the March 1978 baseline DTHR blanket design with the added capability for electrical power production. The blanket concept should be applicable to the U-Pu fuel cycle as well.

A commercial hybrid reactor performance was estimated based on the DTHR results. For the design of an optimum commercial blanket of this type, additional neutronic calculations involving fuel shuffling, equilibrium blanket operating conditions must be performed, economic fuel management scenarios must be developed, and economic evaluations involving symbiotic fusion-fission reactors must be carried out.

This work represents only the second iteration of the conceptual design of a DTHR blanket; consequently, a number of issues important to a detailed blanket design have not yet been evaluated. The most critical issues identified are those of two-phase flow maldistritution, flow instabilities, flow stratification for horizontal radial inlfow of boiling water, fuel rod vibrations, corrosion of clad and structural materials by high quality steam,

fretting and cyclic loads. Approaches to minimizing these problems are discussed and experimental testing with flow mock-ups is recommended. These implications on a commercial blanket design are discussed and critical data needs are identified.

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1.0 SUMMARY

The Demonstration Tokamak Hybrid Reactor (DTHR) was conceived as a near-term fusion engineering test reactor. Its principal mission is to produce a significant amount of fissile fuel while demonstrating the feasibility of the tokamak hybrid reactor concept. This would be accomplished by the integration of critical reactor technologies such as superconducting field coils, plasma exhaust system, breeding blanket, etc. and would employ the simplest, least complex design approach. As a consequence, an important on-going goal of the present study is to make maximum use of proven light water fission reactor technology. This approach is not intended to preclude consideration of more advanced concepts in later stages of the study or the testing of advanced fuel and power conversion technologies in the DTHR.

In line with these objectives, a near-term, fissile breeding blanket concept based on the thorium-uranium (proliferation resistant) fuel cycle was completed as the baseline design in March, 1978[1]. Thorium oxide fuel with zircaloy clad fuel rods, cooled by low temperature and low pressure water was selected. The blanket was dedicated to a demonstration of fissile breeding with no provision for electrical power production potential or tritium breeding. The hybrid blanket did not have any provisions for tritium breeding because tritium breeding would be accomplished in separate blanket modules, particularly blankets for the inner torus region. For the DTHR, upgraded blanket concepts that contain integral tritium breeding could be considered in the future as alternate approaches to be evaluated and compared. A number of economic studies have concluded, however, that economic commercial tokamak hybrid reactors will probably need to be at least self-sufficient in electric power. Since the baseline design indicated a potential for water cooled concepts, DTHR blanket design and analyses were carried out in the latter period of FY 78 with the goal of developing a blanket concept that has improved structural reliability and the capability of electrical power production, i.e. a concept that has greater potential for commercial applications when compared to the baseline blanket concept. The original fuel form and clad material were retained. This entails blanket concepts that are capable of operating at high temperatures and high pressures suitable for efficient power conversion and consistent with LWR operating experience. This topical report discusses the design approach and presents the DTHR blanket concept that evolved. Applications of the concept to a commercial hybrid are also discussed and blanket performance is estimated.

The reference blanket concept was selected following thermal, mechanical and neutronic scoping calculations. The requirement for power conversion entailed high temperature and high pressure blanket coolant operating conditions. The necessity for containing high pressure coolants typically leads to relatively high structural material volume fractions, which have detrimental effects on the neutronic performance because of the close proximity of the structural materials to the fueled region in a fusion blanket. The structural requirements are also directly related to the need to limit the coolant pumping power.

For economical reactor operation, the blanket coolant pumping power must be kept reasonably low. Depending on the state of the coolant and the flow configuration, the pumping power can be reduced by increasing the pressure (to increase the coolant density). However, an increase in the pressure leads to high structural volume fractions. The higher coolant density leads to greater neutron moderation. Preliminary neutronic scoping calculations showed that high water number densities (single phase water or high density steam/water) lead to high neutron moderation. This enhances fissile production, but also induces fissioning of the ^{233}U, causing the net fissile production to decrease with increasing neutron exposure. It became apparent that the water density distribution is another major design parameter in a number of trade-offs that involve power conversion efficiency, coolant pumping power, fissile breeding, and energy multiplication.

Alternate blanket and coolant flow systems were evaluated. The coolant flow systems included moderate to high pressure water, dry steam, two-phase (boiling water in both vertical and radial flow configurations). The latter flow configuration can increase the coolant flow cross-sectional area by a factor of 3 while reducing the coolant flow path length by a factor of $\sim$ 6. The overall effect is a reduction in the pumping power by a factor up to 200 (for the same coolant operating conditions). Blanket configurations considered included rectangular pressure vessels, pressure tubes and rectangular pressure vessels with split cylindrical heads. The results of the evaluation showed the following:

- The maximum neutron exposure, coolant temperature and pressure are limited by the coolant pumping power for <u>vertical flow of single phase water</u> in a structurally fabricable rectangular pressure vessel with wall thickness that are acceptable from the thermal and mechanical stress standpoints.
- <u>Vertical flow of dry steam</u> led to similar limitations. The lower water number densities resulted in reduced neutron moderation, lower power densities and lower flow requirements. However, the coolant pumping power requirements were excessive (greater than the blanket thermal power) for reasonable neutron exposures. A thicker blanket is needed for effective utilization of the high energy neutrons.
- <u>Vertical flow of two-phase (boiling) water in rectangular or square pressure vessels</u> appreciably reduced the coolant pumping power requirements. However, the distribution of water number densities leads to significantly nonuniform axial as well as radial power densities and fissile concentrations, and nonuniform neutron leakage from the back of the blanket. This leads to flow distribution problems. A solution of the flow distribution problem requires radial flow separators and orificing of the parallel flow channels, but this does not solve the latter problem. In general, the neutronic performance is intermediate between the first two cases investigated.

- Although the use of pressure tubes permitted high temperature and high pressure coolant operating conditions (water, dry steam, or two-phase, boiling water) for efficient power conversion and low coolant pumping power requirements, the structural volume fractions required reduced the fuel loading appreciably. In addition, the high void fraction requires a thicker blanket for efficient neutron utilization. The high density coolant leads to excessive moderation of the 14 MeV neutrons. The result is that the neutronic performance becomes relatively poor (relative to the pressure vessel type of blankets).

- Radial two phase coolant flow in a rectangular pressure vessel solved most of the problems delineated above. Radial outflow with single phase water entering from a plenum facing the plasma and exiting as dry steam from the back of the blanket is an attractive flow system from the thermal hydraulic standpoint, because the highest heat transfer coefficient corresponds to the hot rod, while the heat transfer coefficient decreases with decreasing power density. However, the high water number densities at the front of the blanket caused excessive moderation of the 14 MeV neutrons. This led to neutronic performance similar to the low temperature, low pressure water-cooled blanket.

- A rectangular pressure vessel with a split cylindrical head and cooled by radial inflow of two-phase boiling water provided the most favorable neutronic performance while satisfying thermal-mechanical design criteria. The relatively low density wet steam at the front of the blanket retains a relatively hard neutron spectrum which prevents ^{233}U thermal neutron fissioning and thus increases the fissile enrichment. The high density water in the back of the blanket slows down the high energy neutrons to minimize leakage from the blanket and enhance power production. Thermal hydraulic performance was found to be attractive if the blanket exit steam quality is kept at 80% or less. Coolant pumping power is very low, while the DNB ratios* are relatively high. This blanket-coolant flow system with all of its attractive features

*Departure from Nucleate Boiling (flux ratio) was therefore selected as the reference DTHR blanket concept. A cross-section and a side view of the DTHR blanket is shown in Figure 1-1.

The use of this DTHR blanket could be considered in a first generation commercial hybrid. Accordingly, the performance of the reference DTHR in a commercial operation, i.e., in an equilibrium operation mode where refueling and fuel shuffling is carried out once every 2.33 years, was estimated based on the DTHR calculations. The performances are summarized in Table 1-1 for a commercial application with the basic dimensions and plasma performance associated with the reference DTHR system. The DTHR and commercial parameters are compared in Table 1-2. The commercial hybrid characteristics are compared in Table 1-3 with those of typical water power reactors.

The operating conditions were selected to provide enriched fuel rods containing 3% ^{233}U, starting with fertile ^{232}Th. It should be pointed out that this work represents a very preliminary stage in the conceptual design of a DTHR blanket. No attempt was made to optimize the DTHR blanket design. For a commercial blanket, design optimization must be carried out in conjuntion with economic anlayses, which in turn involve the many trade-offs discussed. In addition, fuel shuffling and equilibrium blanket operating conditions must be analyzed in conjunction with specified fuel management scenarios.

The most critical issues that have been identified for the DTHR blanket concept are the potential problems due to radial two-phase flow maldistribution, flow instabilities, flow stratifications, rod vibrations, fretting, and corrosion of stainless steel and zircaloy by steam. Some of the problems may be alleviated by reducing the steam quality throughout the blanket, i.e., lower quality steam at the exit. Experimental studies using flow simulations or electrically heated rods and flow mock-ups are reocmmended. Other design and anlayses needed include studies on tritium diffusion and pick-up in the water coolant. The structural materials data for 316-SS and zircaloy at EOL fast nuetron fluence and experimental study of pulsed thermal and pressure loads on ceramic fuel integrity, fuel clad interactions and stress corrosion must be carried out.

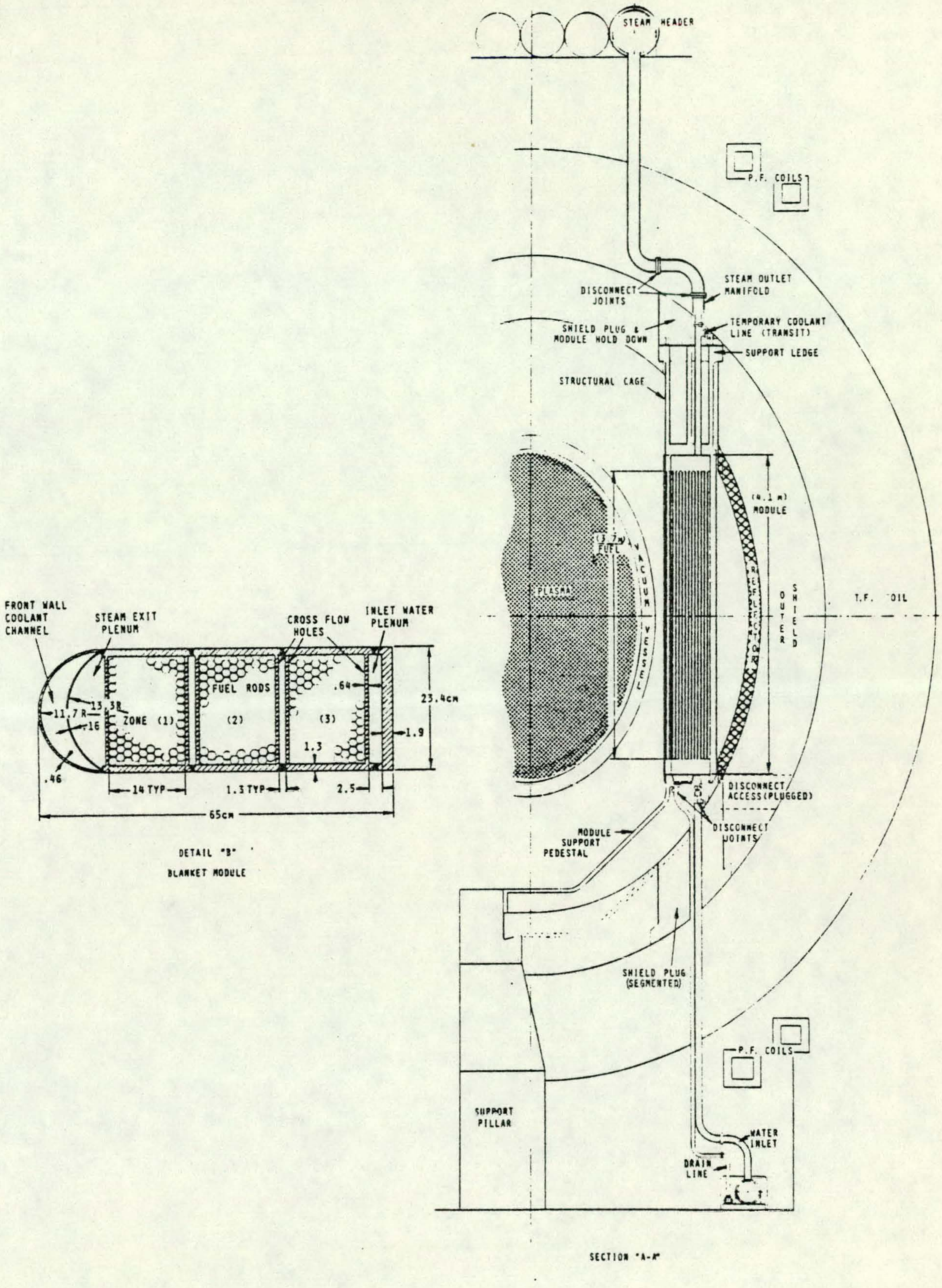


FIGURE 1-1. Cross Section and Side Views of the DTHR Blanket Module.

TABLE 1-1

ESTIMATED COMMERCIAL FERTILE BLANKET FISSILE AND POWER PRODUCTION CAPABILITIES

	DTHR*	COMMERCIAL HYBRID**
BLANKET THICKNESS, cm	40	40
TIME AVERAGED ^{233}U PRODUCTION RATE, kg/yr	165	1200
ACHIEVABLE ^{233}U CONCENTRATION IN Th IN THE FIRST 10 cm %	0.5	3†
AVERAGE BLANKET THERMAL POWER PRODUCTION, MWt	508	4000†
GROSS PLANT THERMAL EFFICIENCY, %	29	29
AVERAGE GROSS BLANKET ELECTRICAL POWER OUTPUT, MWe	147	1160

* Based on 1.2 MW-yr/m^2 neutron exposure (3 years of DTHR operation)
**Estimated equilibrium blanket operating conditions
† Based on 8.4 MW-yr/m^2 cumulative neutron exposure (with forward shuffling, towards plasma, and 7 years of cumulative irradiation prior to removal from the blanket or refueling every 2.33 years).

TABLE 1-2
TOKAMAK PARAMETERS OF THE DTHR AND A COMMERCIAL APPLICATION

	DTHR	EXTRAPOLATION TO COMMERCIAL APPLICATION
MAJOR RADIUS, m	5.2	5.2
MINOR RADIUS, m	1.2	1.2
PLASMA ELONGATION	1.6	1.6
NEUTRON WALL LOADING, MW/m^2	2	2
PLASMA DUTY CYCLE (DC), % ON	50	82
PLANT AVAILABILITY (PA), % AVAIL.	40	70
PLASMA D-T POWER, P_f, MW_t	950	950
PLANT FACTOR, DC × PA	0.20	0.60
INTEGRATED WALL LOADING PER YEAR OF OPERATION, $MW\text{-}Yr/m^2$	0.4	1.2
POLOIDAL BLANKET WALL COVERAGE FRACTION, b_p	0.43	0.67
TOROIDAL BLANKET WALL COVERAGE FRACTION, b_t	0.56	0.87

TABLE 1-3

COMPARISON OF EXTRAPOLATED COMMERCIAL HYBRID BLANKET CHARACTERISTICS WITH THOSE OF TYPICAL POWER REACTORS

	REACTOR TYPE			
	PWR W	BWR (GE)	CANDU	CTHR*
FUEL ROD O.D., cm	0.94	1.25	1.52	1.45
FUEL ROD PITCH, cm	1.25	1.62	1.65	1.60
AVERAGE CORE/BLANKET POWER DENSITY, W/cm^3	104	56	12.4	66
MAXIMUM CORE/BLANKET POWER DENSITY, W/cm^3	249	120	33	174
AVERAGE HEAT FLUX, W/cm^2	68.5	50.3	50	24**
MAXIMUM HEAT FLUX, W/cm^2	183	112	115	183
MINIMUM DNBR (FLUX RATIO)	1.3	1.9	--	1.2
MAXIMUM FUEL TEMPERATURE, °C	1,788	1,829	1,500	1775
SYSTEM PRESSURE, BARS	155	72	89	56
REACTOR THERMAL POWER, MW	3,411	3,579	1,612	4000
GROSS PLANT THERMAL EFFICIENCY	33.7	33.5	31.0	29 †
GROSS ELECTRICAL POWER OUTPUT, MWe	1,150	1,200	500	1160
AVERAGE BURNUP, MWD/T	33,000	27,500	10,000	9690

*Based on 8.4 $MW\text{-}Yr/m^2$ maximum neutron exposure.

**Assumed fuel shuffling.

†Assumes direct steam cycle.

2.0 DESIGN GUIDELINES, GOALS AND APPROACH

2.1 INTRODUCTION

The initial objective of the Demonstration Tokamak Hybrid Reactor (DTHR) blanket design was to develop a state-of-the-art baseline concept that has the capability for breeding a significant amount of fissile fuel[(1)]. This is consistent with the general goal of the DTHR: To demonstrate the successful integration of the major subsystems in a tokamak reactor. In the initial phase of the DTHR program, a near-term, low-cost, fissile breeding blanket was selected as a baseline DTHR concept. The blanket utilized thorium oxide fuel, clad in zircaloy and cooled by low temperature, low pressure water. The blanket was dedicated to a demonstration of fissile breeding with no provision for electrical power production potential. Tritium breeding was considered to be carried out in upgraded or separate blanket modules, particularly ones that can occupy the inner blanket region and outboard of the fissile breeding blanket modules. Tritium breeding was therefore not included in this blanket design study. This blanket topical report deals with the scoping evaluation and evolution of a DTHR blanket that has the potential for significant electrical power production while maintaining the high performance in fissile breeding. The evolution of the concept is discussed.

Commercial performance was estimated based on the predicted performance of the DTHR to assess the potential of the concept; however, fuel shuffling and equilibrium blanket operating conditions were not analyzed in this study.

2.2 DESIGN REQUIREMENTS AND CONSTRAINTS

The basic goals of the DTHR have a number of important ramifications that affect the blanket design. This is best elucidated by comparing the major design goals of the DTHR and a commercial application as summarized in Table 2-1.

It is obvious that an optimum commercial blanket design must provide the lowest cost per unit of fissile material produced or per unit of electricity produced (item 1), depending on whether the hybrid reactor economics is evaluated as an independent system or symbiotically with fission reactors.

In consideration of design goals 1 and 6, it became evident that fuel enrichment goals must be consistent with the economics of reprocessing or with the requirements of light water reactors depending on whether there is reprocessing or no reprocessing.

Design goals 11 and 12 for the commercial application suggested that state-of-the-art fission reactor technologies be adopted to the maximum extent possible. Accordingly, the LWR technologies, which have had the greatest amount of development, coupled with promising hybrid neutronic performance, were retained in the seocnd half of this study for the upgraded DTHR blanket. In addition, LWR fuel rods, zircaloy clad, were specified for the blanket. Pressurized water, boiling water, and steam were considered for blanket cooling.* Rectangular pressure vessels, which maximize the efficient utilization of blanket space and conform to LWR fuel assemblies, were of primary interest. Pressure tube blanket concepts were to be considered in recognition of the need for higher temperature and higher pressure blanket operation for efficient power production. A maximum zircaloy temperature of 300°C was specified based on conservative structural analyses. From the standpoint of neutronic performance, zircaloy pressure vessels were to be given primary consideration, followed by stainless steel.

Previous studies on fusion hybrid reactors and preliminary analyses clearly showed that the blanket power increases steadily with neutron irradiation.[2] It follows that the blanket design must be based on end-of-life operating conditions for applications that do not involve fuel shuffling or the equilibrium condition when the fuel is shuffled. The DTHR modules were assumed to have a three year replacement schedule (1.2 MW-Yr/m^2). This was considered to be long enough for meaningful studies on the effect of neutron irradiation and fissile breeding.

*Helium cooled blanket concepts have already been studied previously under an EPRI funded program[2].

TABLE 2-1

COMPARISON OF MAJOR DESIGN GOALS FOR THE UPGRADED DTHR BLANKET AND A COMMERCIAL BLANKET

DESIGN CONSIDERATIONS	UPGRADED DTHR BLANKET	COMMERICIAL HYBRID
1. Economic	Minimize total capital cost.	Minimize dollars per kw electricity produced or per unit fissile produced for symbiotic systems of the hybrid and client convertor reactors.
2. Fissile Production	> 100 kg/yr	Maximum fissile production within the above constraint.
3. Fissile Concentration	Maximize (Adequate for convincing demonstration)	Consistent with reprocessing and LWR fuel requirements, and consistent with 1.
4. Power Production	Not necessary, potential must be demonstrated	Maximize power production, consistent with 1.
5. Tritium Breeding	Not necessary (can be demonstrated in a separate or an upgraded blanket design).	Maximize tritium breeding, consistent with 1, either in integral or in separate blanket modules.
6. Fuel Cycle	Th-U, potential symbiosis with LWR.	Th-U or U-Pu symbiotic with client convertor or breeder reactors
7. Reprocessing	Not a consideration.	Must be considered in the overall economics of all alternate blanket concepts, consistent with 1.
8. Blanket Refueling and Fuel Shuffling	Capability to be demonstrated	Must be considered, consistent with 1, 2, 3, 4 and 5.
9. Safety	High integrity for fuel clad.	Relative safety of alternate concepts and safety relative to fission reactors must be compared.
10. Waste Management	Consistent with state-of-the-art.	Consistent with fission reactor state-of-the art
11. Impact of and on Materials Resources	Low cost, high availability materials only, consistent with 1.	Must be evaluated in terms of overall economics, consistent with 1.
12. Developmental Risks	Low developmental risks required.	Consideration of concepts based on longer range development scenarios are acceptable only if the economic payoff justifies them.
13. Reliability	Consistent with state-of-the art.	Consistent with commercial fission reactor criteria and with 1.
14. State-of-the art (Overall)	Requires near-term availability of hardware: 1982-1986	Use technology expected to be demonstrated by 1995. First generation commercial deployment.
15. Test Facility	Provide for several blanket configurations and test modules	Not applicable

2.3 DESIGN APPROACH

The basic blanket design approach for an improved DTHR blanket was to select a near-term blanket concept that has the potential for electrical power production. Once a reference concept was selected, design and operating parameters were then modified for accommodation in a DTHR. In accordance with this approach, a major decision had to be made with regard to the vacuum vessel. Preliminary scoping calculations on blanket concepts assumed that there would not be a separate vacuum vessel (separate from the blanket first wall) in the commercial hybrid. However, in the DTHR, a separate vacuum vessel is necessary to provide flexibility and capability for testing different types of blanket modules. In addition, a separate vacuum vessel reduces the tritium contamination (tritium from the plasma) of the blanket water coolant. This design difference has a relatively minor effect on the neutronic performance.

The blanket composition as well as the coolant density has a significant effect on the neutronic performance. This is particularly critical in a hybrid breeder blanket, because, unlike the pressure vessel in a fission reactor, the blanket structural materials (blanket module walls) are in close proximity to the fueled region. In particular, for high temperature, high pressure coolant operating conditions (necessary for electrical power production), structural members can be found within the blanket. Therefore, unlike the preliminary analyses for the conceptual designs of fission reactors, the neutronic analyses for conceptual design of fusion hybrid blankets must be closely integrated with systems, thermal, hydraulic, mechanical and structural design considerations. Accordingly, the systematic design of a hybrid blanket requires design iterations and interactions among neutronic, thermal, and structural-mechanical design and analyses as illustrated in Figure 2-1. The work reported herein is shown enclosed in the dotted box. In general, a sequence of neutronic, thermal and mechanical analyses is necessary to arrive at a final design. However, this approach requires an excessive length of time. In order to minimize the time needed to arrive at an acceptable concept, extensive scoping calculations were performed based on first approximations, provided that the approximations are realistic from the standpoint of overall engineering design considerations.

An initial blanket composition based on dry steam as coolant was developed for neutronic analyses. The power distributions calculated were used as a first approximation in thermal hydraulic scoping calculations to identify feasible and desirable coolant flow schemes. Water density distributions were then calculated for a second iteration neutronic analysis. The power density distributions obtained from the neutronic analyses were used for further thermal-hydraulic and mechanical analyses. This "tandem" iteration process must be continued to arrive at a converged design solution. (A more detailed discussion of this and illustrations are presented in the Appendix.)

Once the coolant flow and general blanket configuration have been established, parametric scoping analyses were carried out to determine the effect of blanket coolant pressure and temperature on power conversion efficiency and blanket structural requirements. The latter in turn determines the blanket composition. This composition as well as the coolant density distribution(s) are needed for neutronic analyses.

A reference blanket concept was selected based on iterative scoping calculations for relatively low neutron exposures (0.4 to 1.2 $MW\text{-}Yr/m^2$). This was then followed by evaluations of the effect of increased neutron exposure on blanket operating parameters. The effects of fuel shuffling, important to the determination of optimum commercial reactor operating conditions, was not studied in this phase of the program. It is anticipated that fuel shuffling and consideration of other means to flatten the radial power distribution in the blanket could lead to a significant performance improvement.

2.4 TOKAMAK DRIVER AND BLANKET PARAMETERS

The tokamak fusion driver and the blanket parameters are listed in Table 2-2. A trimetric of the DTHR based on these parameters is shown in Figure 2-2. The parameters of interest to the blanket design are the neutron wall loading and the plasma duty cycle, which are 2 MW/m^2 and 40%, respectively. For the demonstration blanket, only partial blanket wall coverage was considered, as shown in Table 2-2. The overall blanket wall coverage (in the poloidal and toroidal directions) is 24%. This quantity represents the fraction of the total

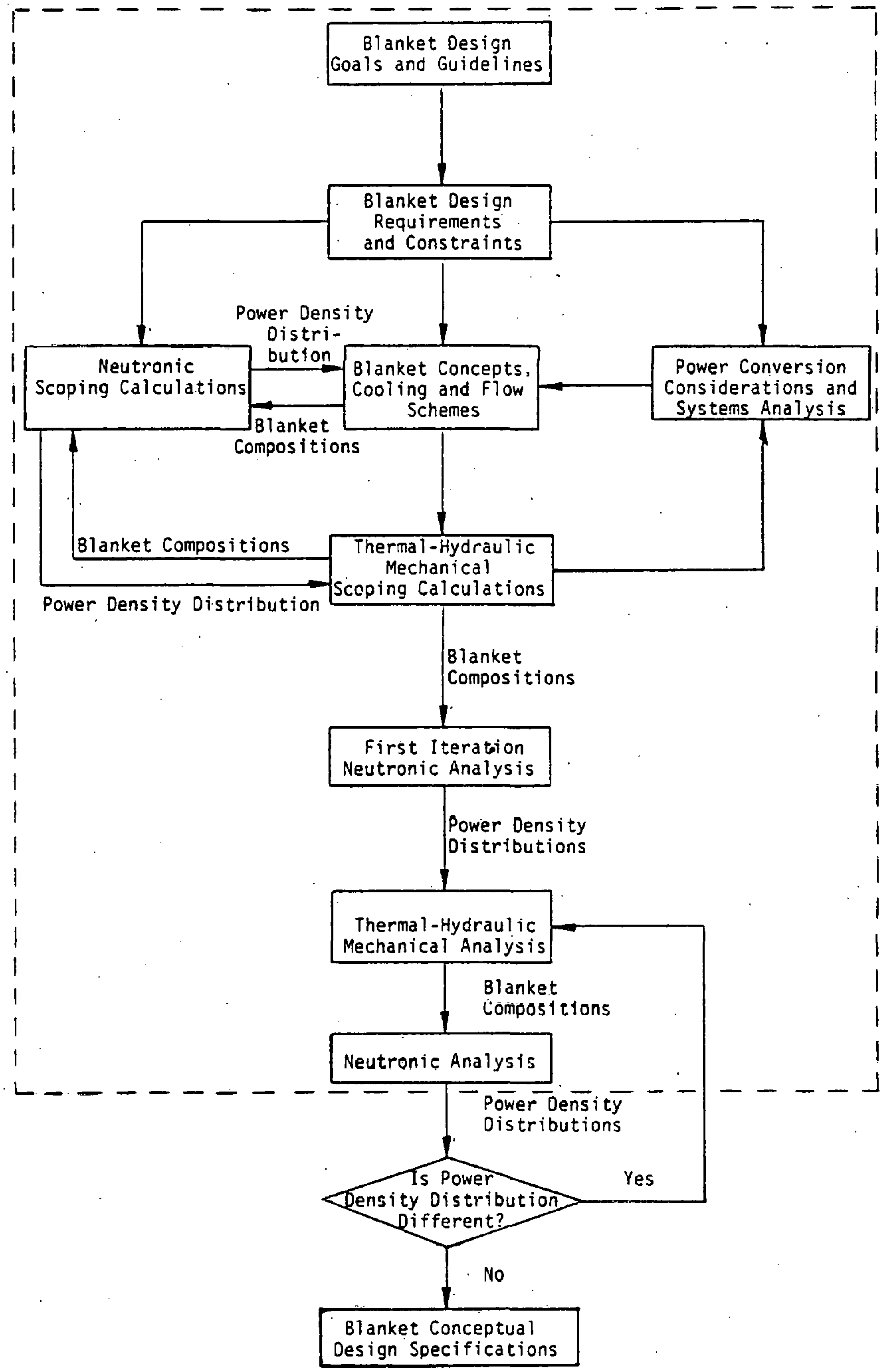


Figure 2-1. Blanket Conceptual Design Approach

fusion neutrons incident on blanket elements. The 2 MW-Yr/m^2 neutron wall loading was selected based on a number of trade-offs associated with assumed plasma beta limits and the peak toroidal magnetic field which could be reasonably attained with present or near-term magnet technology. Parametric trade studies related to this subject are discussed in a separate topical report.*

*"DTHR Plasma Engineering Trade Study," D. A. Sink and G. Gibson, WPFS-TME-79-012, to be published.

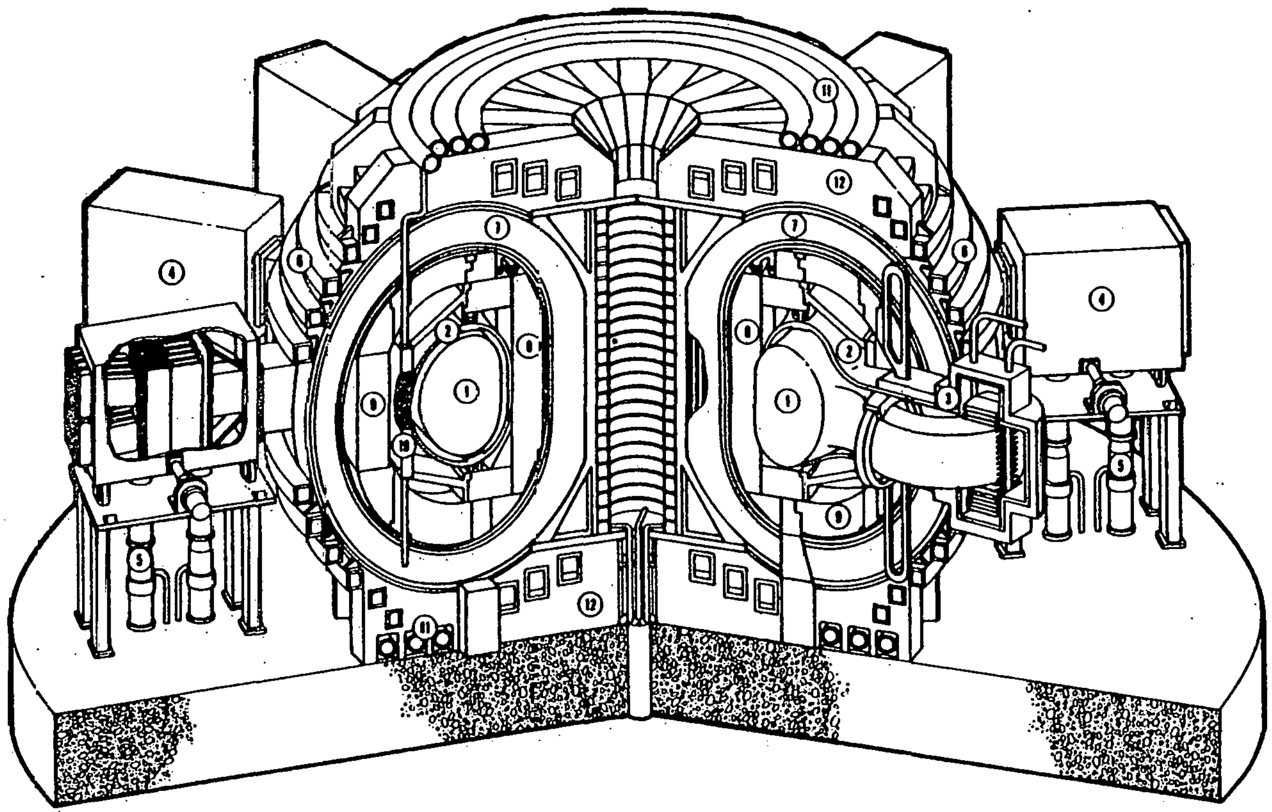
TABLE 2-2

TOKAMAK AND BLANKET PARAMETERS OF THE DTHR

	DTHR
MAJOR RADIUS, m	5.2
MINOR RADIUS, m	1.2
PLASMA ELONGATION	1.6
NEUTRON WALL LOADING, MW/m^2	2
PLASMA DUTY CYCLE (DC), % ON	50
PLANT AVAILABILITY (PA), % AVAIL.	40
PLASMA D-T POWER, P_f, MW_t	950*
PLANT FACTOR, DC × PA	0.20
INTEGRATED WALL LOADING PER YEAR OF OPERATION, $MW\text{-}Yr/m^2$	0.4
POLOIDAL BLANKET WALL COVERAGE FRACTION, b_p	0.43
TOROIDAL BLANKET WALL COVERAGE FRACTION, b_t	0.56

* Final DTHR plasma parameters resulted in a plasma power of 900 MW_t. The analyses were not revised for such a minor change.

DEMONSTRATION TOKAMAK HYBRID REACTOR



LEGEND

1 Plasma
2 Vacuum Vessel
3 Bundle Divertor
4 Neutral Beam Injectors
5 Vacuum Pumps
6 Poloidal Field Coils
7 Toroidal Field Coils
8 Inner Shield
9 Outer Shield
10 Blanket Module
11 Cooling Headers
12 Support Structure

Figure 2-2.

3.0 REFERENCE BLANKET CONCEPT

An improved DTHR blanket concept evolved following considerations of power conversion and following a series of scoping evaluations and parametric analyses of alternate coolants, coolant flow configurations, blanket operating parameters and neutronic performance. This subject is discussed in detail in the Appendix. The sequence of alternate concepts evaluated followed the basic design goals and guidelines already discussed. Basically, the approach was to deviate as little as possible from the state-of-the-art in light water fission reactor technologies. The alternate concepts evaluated are discussed in Section 3-2. The blanket concepts (for a DTHR) that evolved are described and characterized in the following subsections.

3.1 PHYSICAL DESCRIPTION OF THE REFERENCE BLANKET

The reference blanket concept that evolved from the preliminary scoping design calculations is shown in Figure 3-1. The basic characteristics of the rectangular module with a split cylindrical head are listed in Table 3-1. The blanket consists of vertical fuel rods cooled by two-phase (boiling) water in a cross flow (radial inflow) configuration. Typical operating parameters are given in Table 3-2 for 1.2 MW-yr/m^2 of neutron exposure. A schematic diagram of the coolant flow paths is shown in Figure 3-1.

3.1.1 REFERENCE MODULE BUNDLE

The blanket is surrounded by shielding, magnetic coils, piping, structure, etc. so it is important that simplicity be stressed in the overall design of the module. The rectangular geometry allows for blanket bundle removal by basically vertical lifting and lowering motions with offset handling tools working between the TF coils.

The DTHR module bundles are composed of four individual blanket modules which are secured together by a structural framework so that the complete module can be installed and subsequently removed from the reactor as a unit. Each of

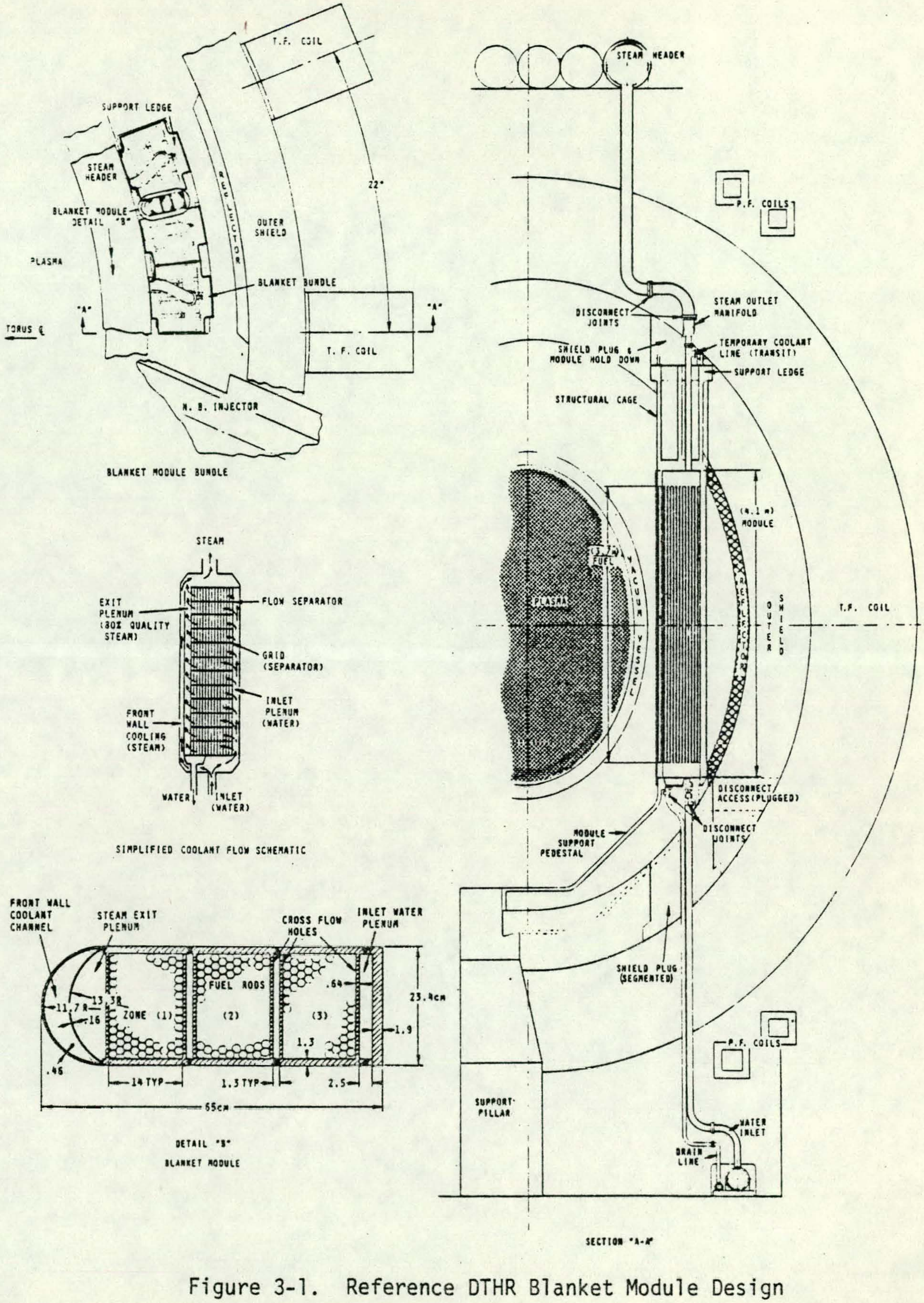


Figure 3-1. Reference DTHR Blanket Module Design

TABLE 3-1
DTHR BLANKET SPECIFICATIONS

WIDTH OF BLANKET MODULE, m	0.23
THICKNESS OF BLANKET MODULE, m	0.65
OVERALL THICKNESS OF FUELED ZONES (3), m	0.40
OVERALL HEIGHT OF BLANKET MODULE, m	5.7
HEIGHT OF FUELED REGION, m	3.7
FUEL ROD DIAMETER, cm	1.45
PITCH TO DIAMETER RATIO	1.1
FUEL CLAD MATERIAL	ZIRCALOY
CLAD THICKNESS, cm	0.065
FUEL FORM	ThO_2
BLANKET COMPOSITION, %	
FUEL	45.4
CLAD (ZIRCALOY)	9.5
COOLANT	30.6
STRUCTURE (S.S. - 316)	13.5
VOID	1.0

TABLE 3-2

REFERENCE DTHR BLANKET OPERATING CONDITIONS

AT 1.2 MW-Yr/m^2 NEUTRON EXPOSURE

AVERAGE BLANKET POWER DENSITY, W/cm^3	15.0
PEAK BLANKET POWER DENSITY, W/cm^3	55
PEAK FUEL ROD POWER DENSITY, W/cm^3	100
MAXIMUM HOT ROD LINEAR POWER, W/cm	165
MAXIMUM WATER MASS VELOCITY, g/cm^2-s	6
INLET COOLANT PRESSURE, BARS	33.3
INLET COOLANT TEMPERATURE, °C	222
OUTLET COOLANT TEMPERATURE, °C	237
INLET WATER SUBCOOLING TEMPERATURE, °C	20
HOT ROD MAXIMUM HEAT FLUX, W/cm^2	36.25
HOT ROD CLAD SURFACE TEMPERATURE, °C	250
HOT ROD MAXIMUM CLAD TEMPERATURE, °C	265
HOT ROD MAXIMUM FUEL TEMPERATURE, °C	655
MINIMUM DNB RATIO (FLUX RATIO) THROUGHOUT THE BLANKET	3.2
COOLANT PRESSURE DROP THROUGH THE BLANKET, BARS	2.4
COOLANT PUMPING POWER TO BLANKET THERMAL POWER RATIO	0.001

the four blanket modules have a rectangular cross-sectional shape in which the surface closest to the plasma is semi-cylindrical (See Figure 3-1). The outside rectangular dimensions are approximately 23 cm wide, 53 cm deep and the radius of the semi-cylinder front section is $\sim$ 12 cm, which results in a total depth of 65 cm. The vertical height of the individual blanket module is 5.7 m, which is made up of 3.7 m of fuel and the remainder being the fuel rod gas plenum, inlet and outlet manifolds, and the upper and lower support structures. When four of these blanket modules are grouped together to form a module bundle, the horizontal cross-sectional space is approximately 81 cm wide by 95 cm deep (this includes the outside structural supporting members).

The coolant inlet flow for each module bundle emanates from a bottom feed water inlet header, 26.7 cm inside diameter, which branches off to each module bundle with a 6.8 cm inside diameter header and then to a 3.4 cm inside diameter line for each blanket module. A top steam exit line of 6.8 cm inside diameter leaves each blanket module and feeds into a 13.2 cm inside diameter header for each module bundle, and then into a 52 cm inside diameter steam header which handles the entire blanket coolant return. Each blanket module also will have a bottom drainage line to collect the condensate and to drain the module bundle prior to removal from the reactor. The coolant feed and discharge lines will be provided with sufficient valving and fittings so that an auxiliary coolant system can be connected if required to remove decay heat while transferring the module from the reactor to a shielded storage container or water pit.

The structural framework which secures the individual blanket modules to form the blanket bundle is made up in the shape of a cage support with a bottom, top and side members which will either be bolted or welded together. The bottom plate will be designed to form a box-type tight enclosure to support the four blanket modules with its lower surface containing a truncated cone shaped socket projection to guide the module bundle into proper alignment for seating. The vertical members between the bottom support and the top plate will be simple rods with the limiting function of structurally connecting the members to form a rigid box-like frame. The top plate will have a built-in lifting member for the movement and handling of the complete module bundle and will have outboard

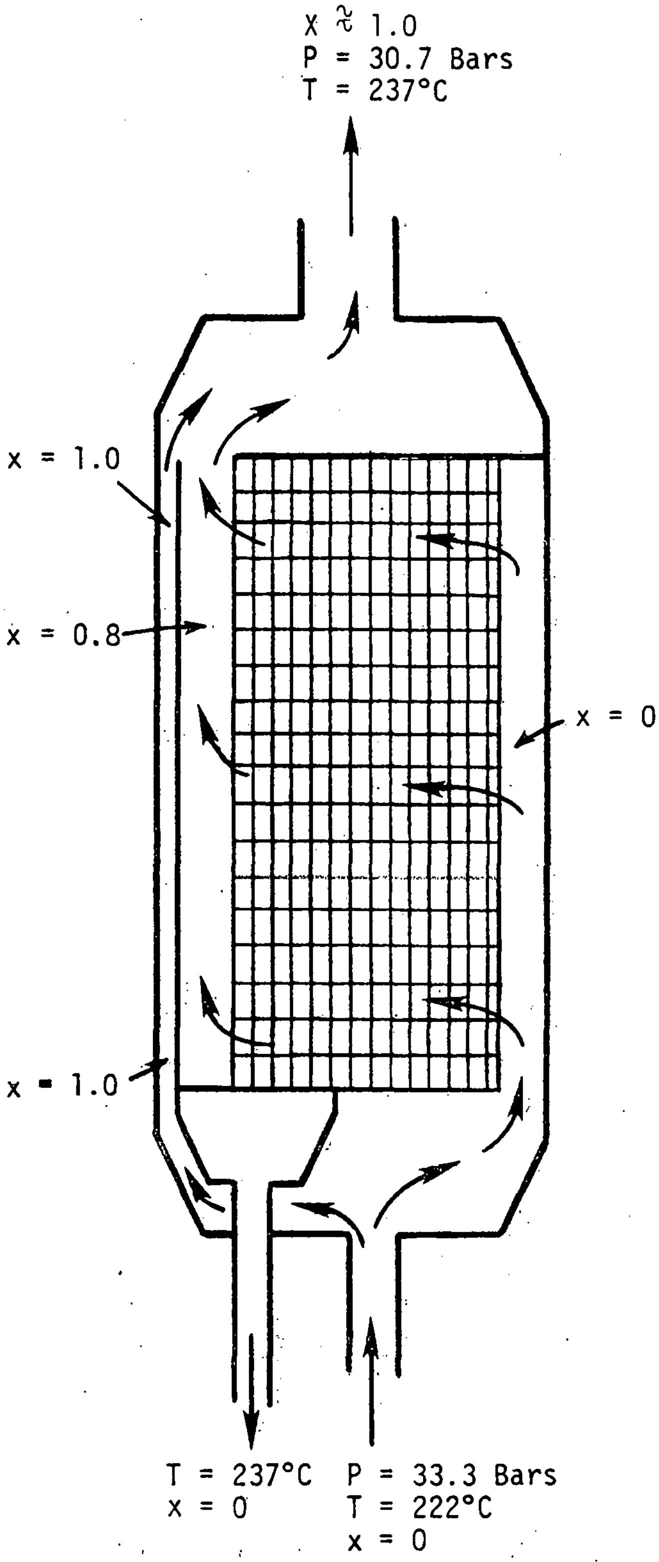


Figure 3-2. Simplified Schematic of Coolant Flow Through the Blanket.

support ledges which will seat in recessed cutouts in the toroidal field coil shielding. The shield plug will sit on top of the module bundle and be secured so as to act as a holddown support and provide sufficient downward force to lock and secure the module bundle in position.

The module bundles will be located in one group of three and four groups of six around the circumference of the torus (see Figure 3-3) with a central module bundle being located in the clear opening between two toroidal field coils and the other two module bundles being located one on each side and slightly under the toroidal field coils.

3.1.2 THE BLANKET ASSEMBLY

The blanket assembly includes supply and return headers for the coolant which can be disconnected remotely from the return and supply piping. The module also has inlet and exit (expansion) plena to accommodate the volumetric change of the two phase water coolant as it flows radially inward through the fueled region. The split cylindrical front wall has a separate parallel cooling system with dry steam as coolant. The coolant streams are fed from the same header (See Figure 3-2). Dry steam for the front wall is obtained by flashing. This is necessary for effective module first wall cooling. The rear reflector has its own cooling system which can be designed to control the bulk temperature in the rear of the blanket. The blanket is divided into three zones so that the separate fuel assemblies can be interchanged for fuel shuffling. The physical boundaries between the fuel rod assemblies of the three zones would be the perforated stiffening ribs. The rectangular pressure vessel split cylindrical front wall permits higher blanket operating pressures while minimizing the first wall thickness and first wall thermal stress. This is of utmost importance because the front wall is subjected to high energy neutron irradiation and helium induced swelling so that its operating temperature must be minimized in order to achieve reasonable wall life. The design offers the potential for operation in a commercial reactor without a separate vacuum vessel wall; in this case the cylindrical surfaces would act as the reactor first wall and would maximize the surface area onto which the surface particle and energy flux and nuclear heat are deposited, thus reducing the overall temperature of the first wall.

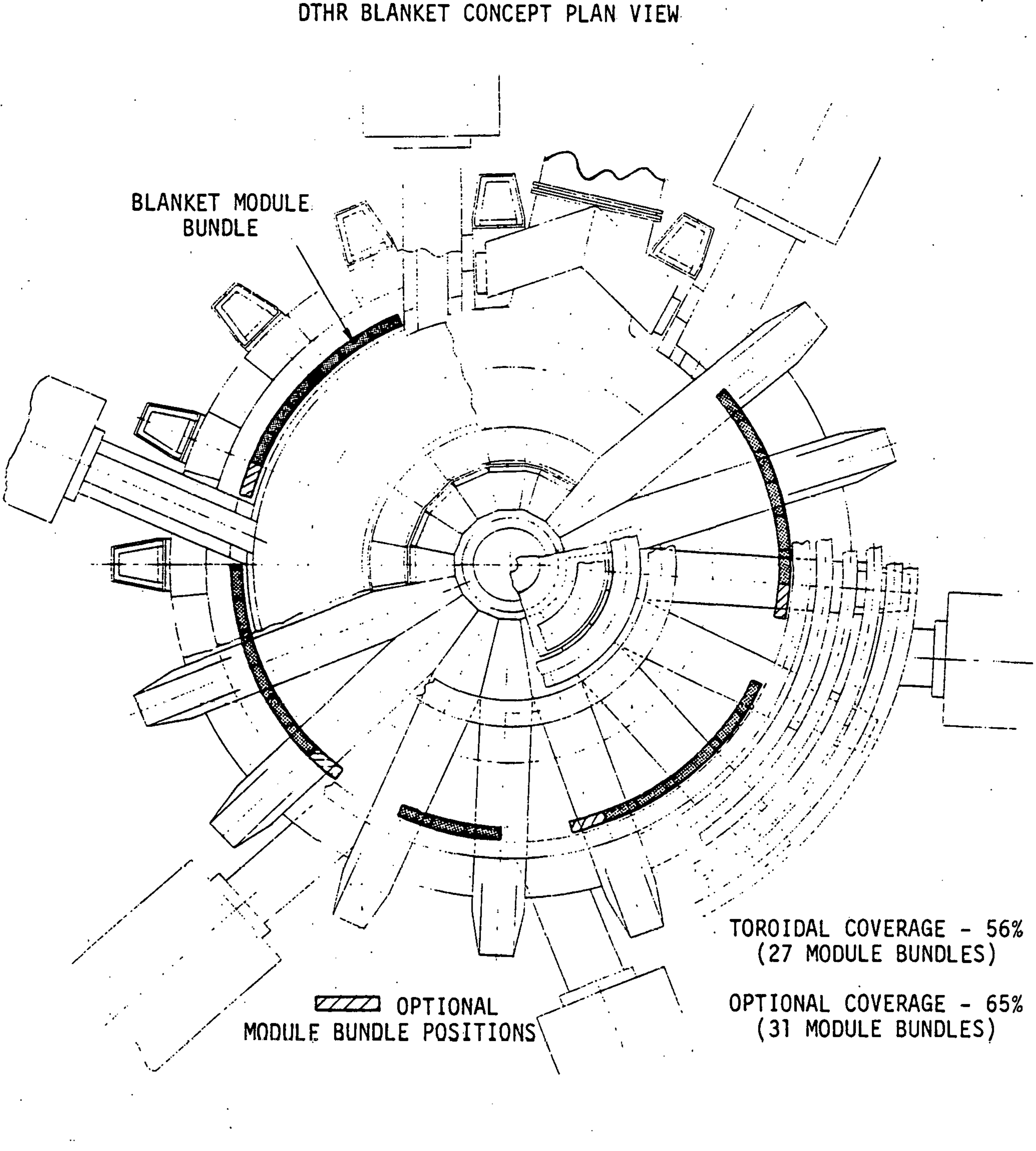


Figure 3-3. DTHR Blanket Concept Plan View.

The reference blanket design has three zones, each zone having a radial depth of 14 cm, allowing 10 rows of fuel rods to be accommodated per zone. The pitch and diameter are 1.6 cm and 1.45 cm, respectively. Overall, each zone has 120 rods with a total of 360 rods per blanket module. Pressurized water (33.3 bars) at a (subcooled inlet) temperature of 242°C is supplied to the back of the module and exiting in the inner cylindrical nose of the module at saturation temperature (254°C).

The exit steam plenum design is constructed from 316 stainless steel with the split cylindrical components welded to form the bullet-like cross-section. The nose is a 180° sector of a ∿ 23 cm O.D. x 0.5 cm wall solution annealed circular tube. Three rectangular tubes in a 20% cold worked condition with outside dimensions of 23 cm wide x 16 cm deep x 1 cm thick wall are provided. The outside surfaces of the wide legs in each duct are undercut 0.5 cm to leave a 0.5 cm lip on each edge. The remaining 0.5 cm wall thickness on both sides is drilled to form a tube sheet with 1.25 cm diameter holes on a 3 cm triangular pitch. The inlet plenum, consisting of a 2.5 cm thick by 23 cm wide 20% CW - 316-SS plate, is machined on one side in the same manner as the rectangular ducts. The nose is welded to the lip formed in the first duct, while the other lip is welded to a lip of the second duct. The third duct is similarly attached. The inlet plenum is welded to the remaining lip of the third duct. This type of welded construction is required to limit the heat affected zone to the lip region without affecting the 20% cold work properties at the critically stressed inside corners of the duct side walls. In this arrangement, three zones with inside dimensions 21 cm wide by 14 cm deep are formed to provide the three fuel rod bundled zones.

3.1.3 FUEL ASSEMBLY DESIGN

Figure 3-4 shows the reference fuel element conceptual design. The fuel rod concept is based on the LWBR blanket fuel rod design except that the rod is longer. The overall outside diameter of the fuel rod is 1.45 cm. The clad thickness is 0.065 cm with a helium gap of 0.008 cm between the thorium oxide (ThO_2) fuel and Zircaloy-4 clad. The pitch to diameter ratio is 1.1. The fuel element concept is identical to that proposed for the low temperature, low pressure DTHR design[1].

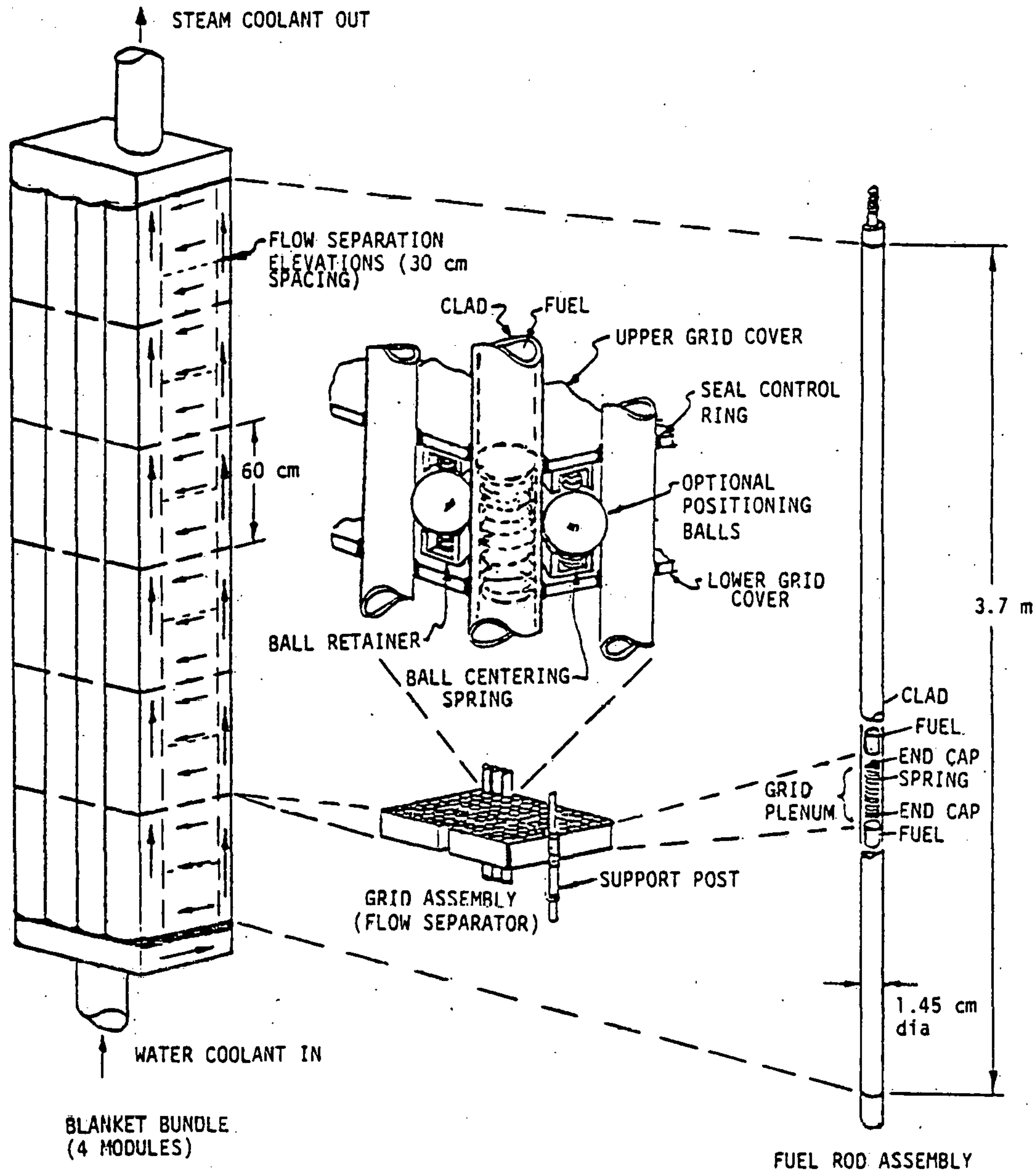


Figure 3-4. DTHR Fuel Rod Assembly - Grid Assembly Conceptual Design.

The fuel rod holding mechanism can be either top or bottom mounted, with the vertically spaced grids acting as flow separators and providing fuel rod stability. The grid assembly concept is similar to the LWBR concept in that it is of light construction to minimize neutron capture. The top and bottom surfaces are covered to prevent coolant from flowing upward or downward between the six controlled passages. The sealing ring at each rod is designed to permit controlled leakage through the grids for cooling the grid components. Use of a positioning ball is optional depending on the movement (both vertically and radially) expected for the rods. Simple leaf springs can be used in lieu of these balls. The location of gas plena at the grids is also optional, depending on the top and bottom rod clearance requirements and the fact that the fuel pellets at the grids could receive less irradiation than adjacent pellets. The coolant enters at the bottom, flows up the outboard inlet plenum, flows radially past the fuel rods and into the steam plenum near the plasma side of the blanket. The front wall cooling flows directly at the bottom (beneath the lowest grid) to the front chamber.

3.1.4 REFUELING PROCEDURE

The following procedure briefly outlines the major steps (not necessarily in the exact sequence) which would be required to remove a module bundle from the DTHR blanket.

1. Attach a lifting device between the crane and the top of the shielding plug (above the center of the three module bundles to be removed).
2. Release any bolts or locking devices securing the shielding block to its adjacent members.
3. Disconnect the cooling lines from the shielding plug and drain the coolant.
4. Using the overhead crane, lift the shielding plug out of its position and move it to a storage stand.
5. Perform steps 1 through 4 to remove the side shield plug to gain access to the bottom of the module blanket.

6. Connect the temporary long flexible coolant lines between the coolant header and the auxiliary coolant fittings on the module bundle.
7. Adjust the valving so that the auxiliary coolant flow is started and the normal coolant flow is stopped.
8. Disconnect the normal (permanent) inlet coolant and the drain lines from the bottom of the module bundle and the steam discharge line from the top of the module bundle.

   Note: It is assumed that provisions will be made to remove the module bundle from the reactor and to place it in a shielded cask or directly into a water pit.

9. Attach the lifting device between the top of the module bundle and the overhead crane.
10. Lift the module bundle vertically upward to clear the bottom support bracket. If the module bundle does not lift free of the bottom support, activate the jacking bolts to free the module bundle.
11. Continue to lift the module bundle vertically until its bottom clears the top of the poloidal field coil just outboard of the module bundle.
12. Move the module bundle over to the entrance to the water channel or the top of a removal cask and lower the module bundle into a storage frame.
13. Disconnect the auxiliary cooling lines from the module bundle. (This may be done just prior to the final lowering of the previous step.)
14. Disconnect the lifting device from the module bundle.
15. If necessary, circulate coolant in the removal cask till the module bundle is discharged into a water pit for temporary storage.
16. Move the crane with the lifting device over the second of the three module bundles to be removed and perform steps 6 through 10 inclusive for the second module bundle. (The lifting device has a built-in offset device so that it clears the toroidal field coil and permits the attachment of the lifting device to the top of the module.)

17. Move the overhead crane so that the module bundle is located in the clear path under the removed shield plug.

18. Repeat steps 11 through 14.

19. Repeat steps 6 through 18 for the third module bundle to be removed.

It is anticipated that shielded personnel carriers, both floor and overhead crane units with external manipulators will be used for the operations. The type of approach and use of shielded personnel carriers are described in an EPRI funded "Remote Maintenance" study(3).

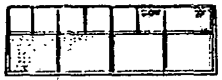
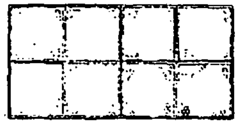
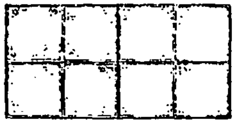
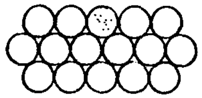
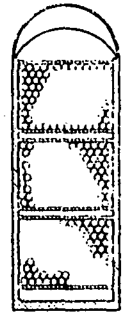
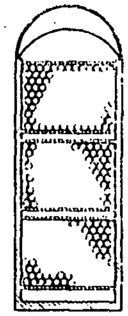
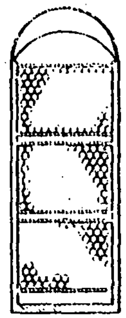
It is estimated that it would require 3 to 4 24-hour working days to perform the above operation at one location. If more than one location is being serviced during the same period, it would be necessary to develop a complete plan for parallel operations to trade-off additional crane and equipment for down time.

3.2 ALTERNATE BLANKET CONCEPTS EVALUATED

The major characteristics, advantages and disadvantages of the alternate blanket concepts evaluated are summarized in Table 3-3. The baseline, low temperature, low pressure, water-cooled blanket concept (developed early in the study program) is included to provide comparisons. The major disadvantage of this concept, as noted previously are: a) that it has no capability for electrical power production and b) if fissile production is a major goal, the neutrons are thermalized too rapidly for optimum fissile production. This concept resulted from the specific goal for the simplest, least complex blanket concept for initial operation in a demonstration blanket. The other alternate concepts all contain capabilities for significant power production, while maintaining capabilities for fissile production.

The second and third concepts require excessive and high coolant pumping powers, respectively. This is due to the fact that only relatively low coolant pressures can be acceptable in the square pressure vessels, while the coolant flow cross-sectional area (in the vertical flow direction) is limited. Even with two-phase boiling water as coolant (concept 3), the maximum neutron exposure allowed is limited by the coolant pumping power requirements so that power and fissile

TABLE 3-3
ALTERNATE BLANKET AND COOLANT FLOW SCHEMES EVALUATED

	1	2	3	4	5	6	7
BLANKET CONCEPT GENERAL DESCRIPTION	RECTANGULAR MODULES BASELINE CONCEPT	SQUARE BLANKET MODULES	SQUARE BLANKET MODULES	CONVENTIONAL PRESSURE TUBE MODULE	SPLIT CYLINDRICAL HEAD BLANKET MODULE	SPLIT CYLINDRICAL HEAD BLANKET MODULE	SPLIT CYLINDRICAL HEAD BLANKET MODULE
BLANKET CROSS-SECTION							
COOLANT	WATER	DRY STEAM	TWO PHASE WATER	SINGLE PHASE WATER	TWO PHASE WATER	TWO PHASE WATER	TWO PHASE WATER
FLOW ORIENTATION	VERTICAL	VERTICAL	VERTICAL	VERTICAL	RADIAL OUTFLOW	RADIAL INFLOW	RADIAL INFLOW
INLET STEAM QUALITY	0.0	1.0	0.0(SATURATED WATER)	0.0	0.0(SUBCOOLED WATER)	0.0(SUBCOOLED WATER)	0.0 (TWO PHASE WATER)
EXIT STEAM QUALITY	0.0	1.0	1.0(SATURATED STEAM)	0.0	1.0(SATURATED STEAM)	1.0(SATURATED STEAM)	0.8 (SUBCOOLED WATER)
MAJOR ADVANTAGES	• LOW TEMPERATURE, LOW PRESSURE, LOW COST • HIGH FUEL LOADINGS • HIGH FUEL/STRUCTURE RATIO • RELATIVELY NEUTRONIC PERFORMANCE • RELIABLE STRUCTURAL DESIGN	• RELIABLE STRUCTURAL DESIGN	• RELIABLE STRUCTURAL DESIGN • RELATIVELY LOW HOT ROD TEMPERATURE	• RELATIVELY HIGH POWER CONVERSION CAPABILITY • REASONABLE COOLANT PUMPING POWER	• RELATIVELY HIGH POWER PRODUCTION • RELATIVELY LOW COOLANT PUMPING POWER • RELATIVELY LOW HOT ROD TEMPERATURES	• RELATIVELY HIGH FISSILE CONCENTRATION • RELATIVELY HIGH FISSILE PRODUCTION • RELATIVELY LOW COOLANT PUMPING POWER	• RELATIVELY HIGH FISSILE CONCENTRATION, • RELATIVELY HIGH FISSILE PRODUCTION • RELATIVELY LOW COOLANT PUMPING POWER
MAJOR DISADVANTAGES	• NO ELECTRICAL POWER PRODUCTION CAPABILITY	• EXCESSIVE COOLANT PUMPING POWER FOR REASONABLE POWER DENSITIES • MAXIMUM NEUTRON EXPOSURE AND POWER DENSITY LIMITED BY COOLANT PUMPING POWER	• RELATIVELY LOW FISSILE PRODUCTION • NON-UNIFORM AXIAL AND RADIAL POWER AND WATER DENSITY DISTRIBUTIONS UNLESS COMPLICATED FLOW SYSTEMS WERE DEVISED • RELATIVELY HIGH PRESSURE DROP, MAXIMUM NEUTRON EXPOSURE LIMITED BY ΔP	• RELATIVELY LOW FUEL VOLUME FRACTIONS • RELATIVELY LOW FUEL TO STRUCTURE RATIOS • RELATIVELY LOW FISSILE PRODUCTION	• RELATIVELY LOW NET FISSILE PRODUCTION DUE TO MODERATION OF 14 MeV NEUTRONS • RELATIVELY LOW FISSILE CONCENTRATIONS	• MAXIMUM NEUTRON EXPOSURE AND POWER DENSITY LIMITED BY HOT ROD TEMPERATURE CONSTRAINTS	• MORE DIFFICULT STRUCTURAL AND MECHANICAL DESIGN

productions are more limited. A solution to these problems is the use of pressure tubes (concept 4), with which high coolant pressures (> 2000 psia) can be utilized to reduce coolant pumping power. The major disadvantage here is the relatively low fuel volume fractions, the relatively high structure to fuel ratios, and the relatively high coolant densities. Table 3-4 provides a comparison of typical blanket compositions in rectangular and pressure tube blanket modules based on the same overall blanket volume. The high structural (SS) volume fraction in the pressure tube blanket are expected to be detrimental to neutronic performance. It should be emphasized that comparisions of blanket compositions should be made on the same bases. The blanket compositions shown in Table 3-4 correspond to the overall blanket, which includes the fuel lattice as well as blanket module structural materials and voids. A high void fraction is detrimental in that for equal utilization of the same amount of fusion neutrons, a thicker blanket would be required, leading to larger TF coils and hence larger reactors and higher reactor costs. This can be contrasted with compositions (frequently reported in the literature) that are based on the fuel lattice only, as shown in Table 3-5. Greenspan et al.[4] performed extensive neutronic analyses on fusion driven, water cooled hybrid breeder blankets. They showed that such a system has potential for power production as well as fissile breeding. Their analysis, however, was based simply on a unit fuel lattice cell, consisting of fuel, clad and water coolant only. They recognized that "realistic water-cooled blankets will have to be either of a pressure vessel or of a pressure tube design and that either approach will impair the neutron economy of the blankets, causing reduced energy multiplication and reduced fissile breeding." Clearly, significant performance differences, particularly in terms of the neutronic performance obtained, can be expected from the two different sets of compositions.

An advanced pressure tube concept, with potential for significantly improved fuel loadings, was also considered. This consisted of co-extruded metallic thorium-zircaloy clad fuel rods (with central coolant channels) of the type developed at BNWL and SRL[5]. Analysis showed that a blanket containing fuel rods of this type can have fuel volume fractions as high as 38%. However, the

TABLE 3-4
COMPARISON OF BLANKET COMPOSITIONS
BASED ON THE OVERALL BLANKET
(MODULE STRUCTURES, FUEL, CLAD, COOLANT AND VOIDS)

BLANKET MODULE CONCEPT	RECTANGULAR PRESSURE VESSEL	CONVENTIONAL PRESSURE TUBES
GENERAL CONFIGURATION	FUEL RODS IN RECTANGULAR PRESSURE VESSEL	19 FUEL RODS/ PRESSURE TUBE SEVERAL PRESSURE TUBES PER MODULE.
FUEL VOLUME FRACTION	0.45	0.27
COOLANT VOLUME FRACTION	0.31	0.25
ZIRCALOY CLAD VOLUME FRACTION	0.10	0.06
S.S. STRUCTURE VOLUME FRACTION	0.13	0.23
VOID VOLUME FRACTION	0.01	0.19
FUEL/S.S. RATIO	3.14	1.17

TABLE 3-5

COMPARISON OF BLANKET COMPOSITIONS

BASED ON THE FUEL LATTICE ONLY
(FUEL, CLAD, COOLANT ONLY)

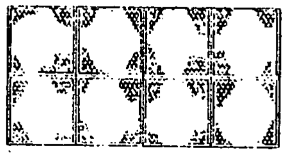
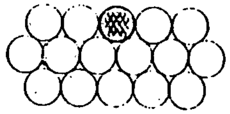
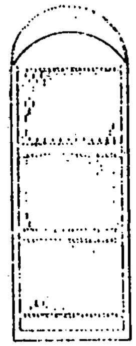
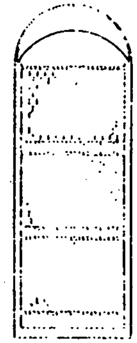
BLANKET MODULE CONCEPT	RECTANGULAR PRESSURE VESSEL	CONVENTIONAL PRESSURE TUBES
GENERAL CONFIGURATION	FUEL RODS IN RECTANGULAR BLANKET MODULES (PRESSURE VESSEL)	19 FUEL RODS/PRESSURE TUBES. SEVERAL PRESSURE TUBES PER MODULE.
	COMPOSITION OF FUEL LATTICE ONLY	COMPOSITION OF FUEL LATTICE ONLY
FUEL VOLUME FRACTION	0.52	0.47
COOLANT VOLUME FRACTION	0.36	0.43
ZIRCALOY CLAD VOLUME FRACTION	0.11	0.10
S.S. STRUCTURE VOLUME FRACTION	0.0	0.0
VOID VOLUME FRACTION	0.01	0.0

appreciably lower heat transfer surface area to volume ratios and flow cross-sectional area for this concept plus the fact that the peak rod temperature occurs at the outer clad results in low coolant temperatures and excessive coolant flow rates/pressure drops/coolant pumping power (see the Appendix for further discussion). In order to contain the pressure from fuel swelling, the inner and outer concentric clads must be thickened appreciably even if zircaloy were replaced with stainless steel. The result is that a realistic fuel rod design of this type would lead to significantly reduced fuel loadings and to high stainless steel fractions. This fact, coupled with the relatively poorly developed technology for co-extruded fuel rods (compared to LWR fuel rods), suggested that the concept not be considered for the DTHR.

The problem of containing high coolant pressures and maintaining reasonable coolant pumping power within design temperature limitations can be substantially alleviated by considering radial flow of two-phase boiling water coolant (concepts 5, 6, 7,). This flow system results in low coolant pumping powers even at moderate coolant pressures. This is due to the higher latent heat of vaporization for water, higher coolant flow cross-sectional area and the shorter coolant flow paths. With moderate pressures, the structural volume fraction in the blanket can be reduced. Three different sets of coolant flow/operating conditions were investigated as indicated in Table 3-3. Radial coolant outflow is ideal from the standpoint of optimum blanket thermal-hydraulic operations. The high density water in the plenum at the front of the blanket moderates the 14 MeV neutrons to such an extent that high net fissile enrichments cannot be attained. Blanket power production is enhanced because of high fissile burnup. In the interest of maximizing fissile enrichment and net fissile production, radial inflow of boiling water coolant (concepts 6 and 7) was investigated next. Concept 6 had 100% quality, dry steam exiting from the blanket (to minimize steam separation problems/requirements) so that the steam may be used directly in a steam generator. However, the heat transfer coefficients associated with high quality steam (at the exit of the module) are relatively low and are associated with the hot rod in radial outflow. In addition, 100% dry steam at the exit leads to the deposit of dissolved chemicals in the blanket and cause corrosion problems. Accordingly, the exit steam quality was lowered to 80% (concept 7). Corrosion by steam of this quality remains a problem, however.

Based on the iterative thermal-hydraulic, structural, and neutronic scoping calculations, the viable blanket-coolant systems were identified. These systems and their unique characteristics are tabulated in Table 3-6. It is clear from the table that the first system, utilizing rectangular pressure vessels and low temperature, low pressure water, has limited potential in a commercial device because of its inability to produce electric power. Although the pressure tube concept (system 2) has higher temperature and higher pressure capabilities, the fuel volume fraction and fuel to stainless steel (structure) ratios are low. The high pressure water coolant leads to a highly thermalized blanket in which parasitic absorptions in stainless steel becomes more important. There is the possibility that stainless steel pressure tubes can be replaced by zircaloy pressure tubes but this is not "state-of-the-art", although it may be developed for near-term application. Vertical flow of two-phase boiling water is feasible. However, the maximum neutron exposure allowed is limited by the pumping power required. A major disadvantage, however, is the significant neutron moderation encountered. This enhances power production at the expense of fissile production with widely varying power densities with irradiation time. The nonuniform vertical power density distributions present complexities in neutronic as well as thermal hydraulic designs. The last two systems with radial flow of boiling water are both attractive, depending on whether power or fissile production is emphasized. With radial outflow of boiling water (high density water in the front), the high energy plasma fusion neutrons are slowed down significantly before they can interact with the blanket fertile material. This degrades the fissile fuel production and fuel enrichment, but could enhance the power production from fission reactions. However, the tritium breeding potential is increased (based on neutron leakage to the back of the blanket). With radial inflow of boiling water, the opposite effects are seen. A final selection of a boiling water flow scheme for a commercial application will depend on overall economic analysis. Based on the goals of the DTHR, the blanket/coolant flow system with radial inflow appeared to be more appropriate. Accordingly, that concept was selected for the reference blanket which is described in more detail in the following sections.

TABLE 3-6
COMPARISON OF VIABLE BLANKET COOLANT FLOW SYSTEMS

	RECTANGULAR PRESSURE VESSEL	CONVENTIONAL PRESSURE TUBES		SPLIT CYLINDRICAL HEAD	SPLIT CYLINDRICAL HEAD
COOLANT	LOW TEMP, LOW PRESSURE WATER	HIGH TEMPERATURE HIGH PRESSURE WATER	MODERATE T/P TWO PHASE BOILING WATER	MODERATE T/P TWO PHASE BOILING WATER	MODERATE T/P TWO PHASE BOILING WATER
FLOW ORIENTATION	VERTICAL	VERTICAL	VERTICAL	RADIAL OUTFLOW	RADIAL OUTFLOW
BLANKET THICKNESS, cm	25	40	40	40	40
TYPICAL OVERALL BLANKET MODULE COMPOSITION, %					
FUEL	57	27	45	45	45
CLAD (ZIRALOY)	13	6	10	10	10
COOLANT	24	25	31	31	31
S.S. (STRUCTURES)		23	13	13	13
VOID	1	19	1	1	1
RELATIVE ELECTRIC POWER PRODUCING CAPABILITY	NONE	MODERATE TO LOW (LOW FUEL LOADING, FUEL TO STRUCTURE RATIO)	HIGH (PARTIALLY THERMALIZED SPECTRUM)	HIGH (HIGHLY THERMALIZED SPECTRUM)	MODERATE (INTERMEDIATE SPECTRUM)
RELATIVE NET FISSILE PRODUCTION	MODERATE (HIGHLY THERMALIZED SPECTRUM)	MODERATE TO LOW (LOW FUEL LOADING, LOW FUEL TO STRUCTURE RATIO)	LOW	MODERATE (HIGHLY THERMALIZED SPECTURM)	HIGH (FAST SPECTRUM UP FRONT)
RELATIVE TRITIUM BREEDING POTENTIAL (INDIRECT BY NEUTRON LEAKAGE THROUGH THE BACK)	LOW	LOW	LOW	HIGH	MODERATE

4.0 DTHR BLANKET ANALYSES

Following the selection of a reference DTHR blanket concept, more detailed neutronic, structural, thermal-hydraulic, and systems analyses were performed to evaluate the operating performance and adequacy of the concept. The results are discussed in this section.

4.1 NEUTRONIC ANALYSIS

The blanket neutronic performance was evaluated using the coupled neutron transport burn-up code system ANISN-CINDER-HIC. The neutron transport was calculated with the one-dimensional discrete ordinates code ANISN[6] using an S_8 treatment of the angular quadrature and a P_3 approximation to the anisotropic scattering. The isotope burn-up and depletion were computed using a combination of the CINDER[7] and HIC[8] codes which were appended[9] to the ANISN code. For each time step in the burn-up calculation, ANISN supplies collapsed five (5) group fluxes and cross sections to the CINDER-HIC module. The burn-up and isotope depletion and transmutation are then calculated using a library of five group cross sections and decay constants, and new isotope number densities are returned to ANISN for the next time step. In this way, the program cycles through the various time steps requested, computing the isotope number densities at each step. Using a library containing the fission product yield and cross section data, the code also calculates a lumped fission product concentration and cross section which is then used in the ANISN calculation to account for parasitic losses from the fission products.

A 27 neutron group cross section set was used for the ANISN transport calculations. It was prepared by collapsing the 171 group CTR processed multi-group cross section library[10] over an appropriate flux spectrum. Resonance effects in the ^{232}Th were accounted for by using the self-shielding capabilities of the library.

The one-dimensional cylindrical geometry model used to assess the nuclear performance of the blanket is shown schematically in Figure 4-1. This type of

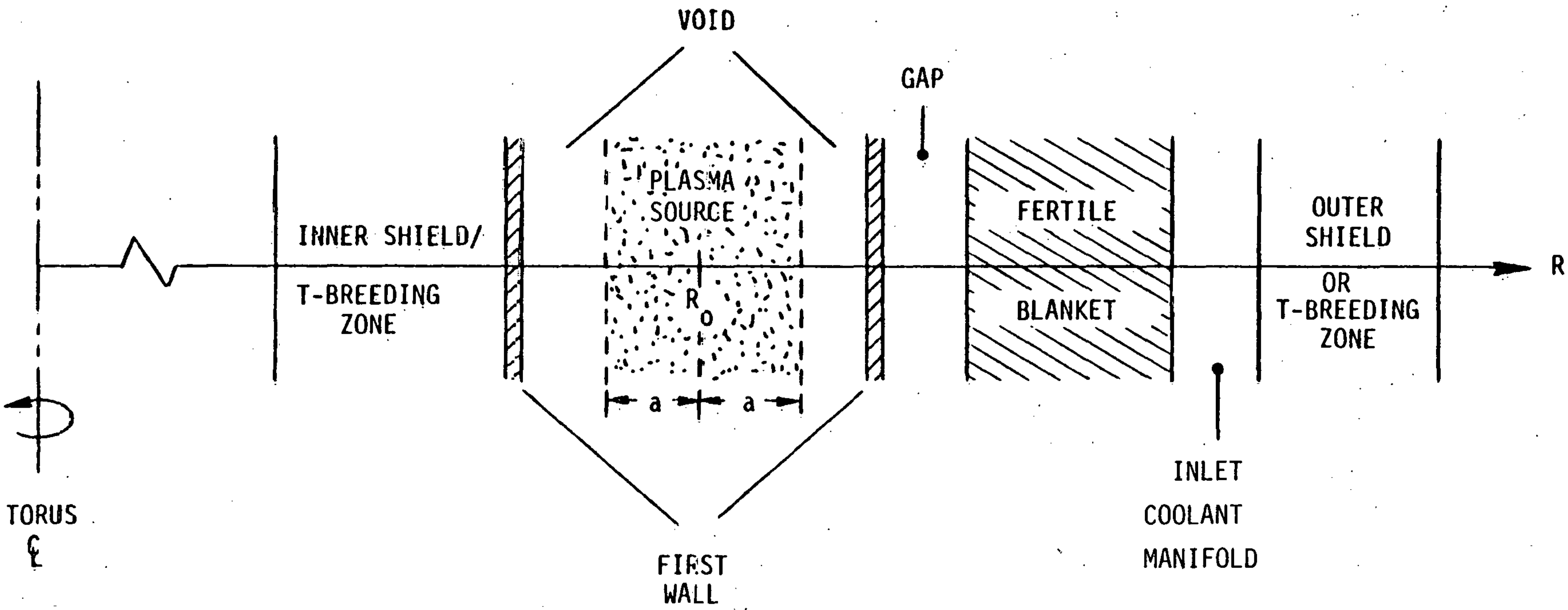


Figure 4-1. Schematic Diagram of the One-Dimensional Cylindrical Geometry Used in the Neutronics Analysis.

TABLE 4-1

TOKAMAK PARAMETERS OF THE DTHR AND A COMMERCIAL APPLICATION

	DTHR	COMMERCIAL HYBRID
MAJOR RADIUS, m	5.2	5.2
MINOR RADIUS, m	1.2	1.2
PLASMA ELONGATION	1.6	1.6
NEUTRON WALL LOADING, MW/m^2	2	2
PLASMA DUTY CYCLE (DC), % ON	50	82
PLANT AVAILABILITY (PA), % AVAIL.	40	70
PLASMA D-T POWER, P_f, MW_t	950	950
PLANT FACTOR, DC × PA	0.20	0.60
INTEGRATED WALL LOADING PER YEAR OF OPERATION, $MW\text{-}Yr/m^2$	0.4	1.2
POLOIDAL BLANKET WALL COVERAGE FRACTION, b_p	0.43	0.67
TOROIDAL BLANKET WALL COVERAGE FRACTION, b_t	0.56	0.87

representation, with the cylinder axis along the centerline of the torus, was selected as the most appropriate for the assumed tokamak hybrid design in which the ThO_2 blanket is placed only on the outer part of the torus. Due mainly to accessibility limitations and maintenance requirements, the inner portion of the torus is used only for tritium breeding and radiation shielding. The isotropic 14 MeV D-T fusion neutron source is distributed uniformly throughout the plasma region. The plasma is surrounded by a void (scrape-off) layer and then by the vacuum vessel/first wall. For these calculations, the tokamak major radius R_0 was 5.2 m, the plasma minor radius was 1.2 m, the scrape-off layer thickness was 0.2 m, and the first wall was a 0.4 cm thick layer of stainless steel.

The most important tokamak parameters with regard to the nuclear analysis of the blanket performance are assumed to have the values shown in Table 4-1. As shown in the table, one of the main differences assumed between the DTHR and a commercial hybrid is in the plant availability and plasma duty cycle. This results in a factor of three increase in the integrated wall loading achievable on one year's time of operation, from 0.4 MW-Yr m^{-2} in the DTHR to 1.2 MW-Yr m^{-2} in the commercial application. It is the value of the plant factor averaged wall loading which is needed to calculate the fusion neutron source strength used in the ANISN-CINDER-HIC calculations.

There are also differences in the assumed blanket wall coverage fractions between the DTHR and the commercial hybrid, as shown in Table 4-1. The value for the poloidal coverage factor, b_p, of 0.43 in the DTHR represents the approximate fraction of the plasma fusion neutrons intercepted by a vertical blanket module of 3.7 m long active fuel length. The commercial reactor uses a value of $b_p = 0.67$, which represents blanket coverage on the outer one-half of the vacuum vessel poloidal surface. The toroidal coverage factor, b_t, which represents the fraction of the tokamak toroidal circumference which is covered by blanket modules, is assumed to increase from 0.56 in the DTHR to 0.87 in the commercial reactor. The value of 0.56 assumes blanket modlues are only located between 9 of the 16 TF coils, while the value of 0.87 represents placing blanket modules in all areas except those needed for the neutral beam injectors and the bundle divertor chamber.

As shown in Figure 3-1, it is assumed that for the DTHR, the fertile blanket is placed only on the outer part of the torus, with shield regions behind the blanket and on the inner part of the torus. Although lithium bearing materials are not assumed to be in these zones, it is possible to estimate the potential tritium breeding ratio of these zones by computing the number of neutrons absorbed in these regions per fusion neutron.

In order to characterize the nuclear performance of the blanket, the following parameters are defined:

$$F \equiv \frac{\text{net number of fissile } (^{233}\text{U}) \text{ atoms produced}}{\text{incident fusion neutron}}$$

$$f \equiv \frac{\text{number of fission reactions in the blanket}}{\text{incident fusion neutron}}$$

$$M \equiv \text{blanket D-T fusion neutron energy multiplication factor}$$

and

$$M \equiv \frac{14.1 + 200 \times f}{14.1} = 1.0 + 14.2 \times f.$$

Note that these values of F and f are defined as the number of reactions per fusion neutron born in the plasma. For the vertical, cylindrical geometry used in the neutronics calculation with a blanket on the outer part of the torus only, not all of the fusion neutrons directly strike the blanket. However, the macroscopic performance parameters such as power and fuel production are accounted for by using the blanket coverage factors b_p and b_t.

The instantaneous blanket thermal power production P_B, is calculated from

$$P_B = P_n \times b_p \times b_t \, [1.389 \times M - 0.389] \qquad \text{MWt}$$

where $P_n \equiv$ plasma fusion neutron power.

The hybrid fuel production G of ^{233}U is computed from

$$G = 7.535 \times F \times P_n \times PA \times DC \times b_t \times b_p \qquad \frac{\text{kg } ^{233}\text{U}}{\text{year}}$$

The material compositions used to calculate the performance of the reference blanket in the DTHR are shown in Table 4-2. For calculational purposes, the blanket was divided into three zones with thicknesses of 10, 14, and 15 cm. An average water coolant density was used in each zone as determined from an overall water density distribution shown in Figure 4-2, which shows the radial distribution of the coolant density. This density distribution is beneficial to the neutronic performance of the blanket since it allows a low density coolant at the front of the blanket to increase the fusion neutron multiplication and a higher density coolant at the rear of the blanket to slow down and reflect the neutrons. A stainless steel vacuum vessel of cumulative thickness 0.37 cm, helium cooled, was adopted in the calculations for the DTHR blanket. This assumption was based on a preliminary vacuum vessel concept developed as a part of the DTHR program. Also, the blanket inlet plenum was modeled as a 10 cm thick zone of low density coolants with a steel volume fraction of 0.083.

The performance of this blanket at beginning-of-life (BOL) and after three years of exposure in the DTHR is shown in Table 4-3. These results indicate that fissile production has decreased by about 6% after three years relative to BOL, giving an average production rate of ∿ 165 kg ^{233}U per year. The power produced in the blanket has increased by ∿ 80% due to increased ^{233}U fissioning, but the total power including that released in the shields has increased by only about 25%. Note that after three yaars the average enrichment of ^{233}U in Th is 0.36% while the peak is 0.51%. The typical enrichment required for PWR startup is ∿ 3%. The potential tritium breeding ratio in these blankets, as measured by the number of neutron absorptions in the shield regions is ≈ 0.6. This increases with irradiation because of ^{233}U fissioning. The performance of a similar two-phase water cooled blanket module under a much longer exposure period is discussed in Section 4.5 of this report, where the prospects for extrapolation to commercial reactor performance are examined.

4.1.1 RADIATION DAMAGE PARAMETERS

Besides providing for the demonstration of breeding significant quantities of ^{233}U in the DTHR, the boiling-water cooled blanket modules also afford an opportunity to study the effects of radiation damage on the materials properties

TABLE 4-2
REFERENCE DTHR BLANKET MODULE
ASSUMED IN THE NUCLEAR CALCULATIONS

THICKNESS, cm	40
ThO_2 VOLUME FRACTION	0.454
COOLANT VOLUME FRACTION	0.306
ZIRC VOLUME FRACTION	0.095
SS VOLUME FRACTION	0.135
GAP (VOID) VOLUME FRACTION	0.010
COOLANT DENSITY, $\frac{gms}{cm^3}$ $\left(\frac{lb}{ft^3}\right)$	
FIRST ZONE (10 cm)	0.032 (2.0)
SECOND ZONE (15 cm)	0.080 (5.0)
THIRD ZONE (15 cm)	0.513 (32.0)
VACUUM VESSEL THICKNESS, cm	0.37
BLANKET EXIT PLENUM	
THICKNESS, cm	10.0
SS VOLUME FRACTION	0.083
COOLANT VOLUME FRACTION	0.700
COOLANT DENSITY, $\frac{gm}{cm^3}$ $\left(\frac{lb}{ft^3}\right)$	0.021 (1.31)

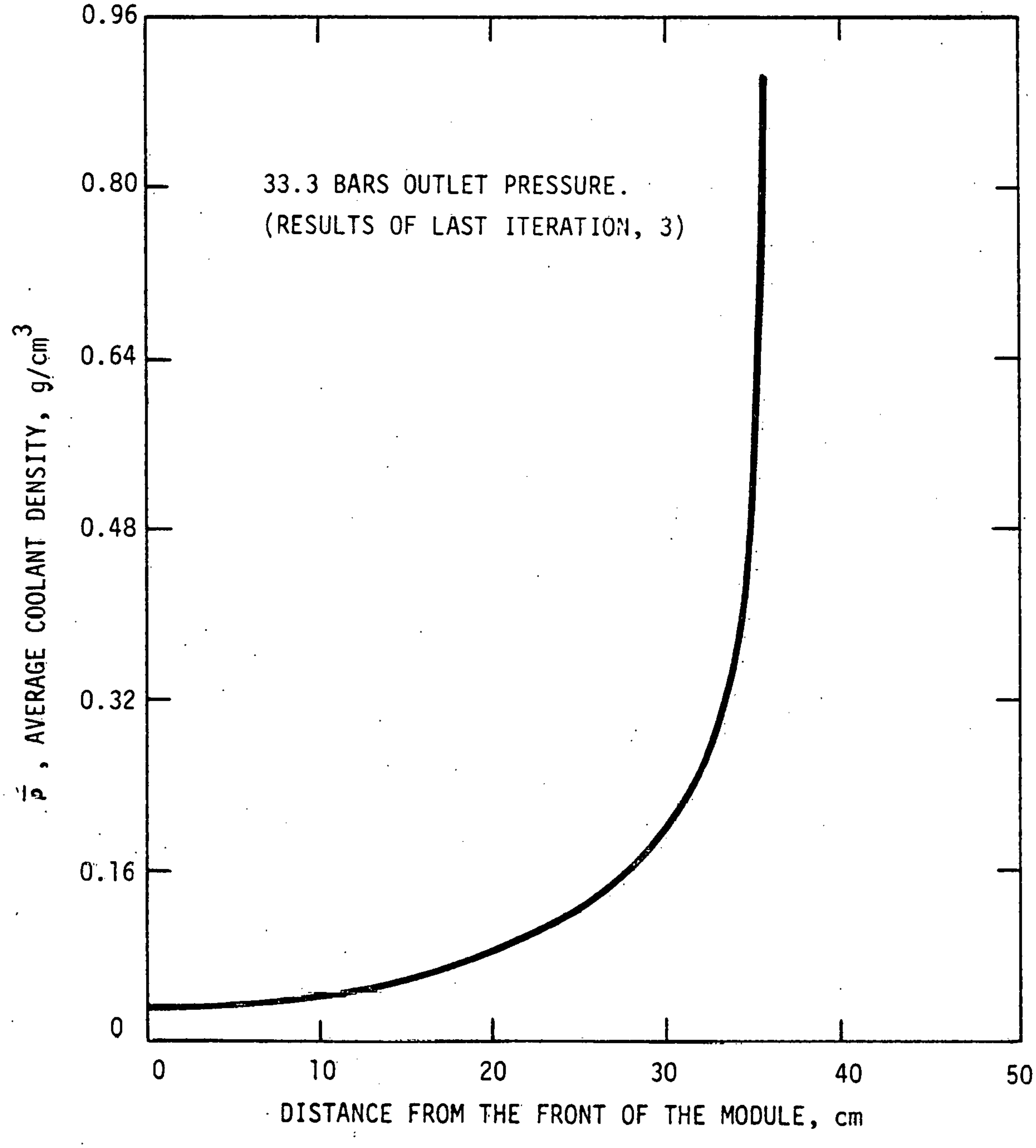


Figure 4-2. Radial Coolant Density Distribution for Radial Inflow of Boiling Water.

TABLE 4-3

PERFORMANCE OF THE DTHR REFERENCE BOILING WATER COOLED ThO_2 BLANKET

	AT BOL	AFTER 3 YEARS
EXPOSURE, $\frac{\text{MW-YRS}}{m^2}$	—	1.2
F, $\frac{\text{NET NO. } ^{233}\text{U ATOMS}}{\text{INCIDENT FUSION NEUTRON}}$	0.617	0.579
M, ENERGY MULTIPLICATION FACTOR	1.63	2.72
ENRICHMENT, w/o ^{233}U IN Th		
FIRST ZONE (10 cm)	—	0.51
SECOND ZONE (15 cm)	—	0.38
THIRD ZONE (15 cm)	—	0.25
BLANKET AVERAGE	—	0.36
SHIELD ABSORPTIONS, $\frac{\text{NEUTRONS}}{\text{INCIDENT FUSION NEUTRON}}$	0.596	0.643
G, $\frac{\text{kg } ^{233}\text{U}}{\text{YEAR}}$ PRODUCTION RATE	170	160
CUMULATIVE ^{233}U, kg	0	495
PEAK BLANKET POWER, P_B, MWt	342	619
AVERAGE BLANKET POWER[(a)], MWt	280	508

(a) At a plasma duty cycle factor of 0.82 for short test periods.

TABLE 4-4

RADIATION DAMAGE PARAMETERS IN THE FRONT WALL OF THE DTHR BLANKET MODULE

FRONT WALL MATERIAL	316 STAINLESS STEEL
EXPOSURE (∿3 YEARS)	1.2 $MW\text{-}Yr/m^2$
NEUTRON FLUX (E > 0.1 MeV)	4.41×10^{14} cm^{-2} sec^{-1}
NEUTRON FLUENCE (E > 0.1 MeV)	1.36×10^{22} cm^{-2}
ATOMIC DISPLACEMENTS	20.3 DPA
HELIUM PRODUCTION	257 appm He

of the stainless steel structure. The estimated radiation damage parameters that will occur in the stainless steel front wall of the blanket module after three years of operation in the DTHR are presented in Table 4-4. These values should create appreciable effects in the material properties in a large test volume, hence providing the DTHR with the added capability of serving as a materials test reactor.

For the analysis of the nuclear performance of the DTHR blanket, a 15 cm thick reflector zone with a composition consisting of 35% water and 65% SS was assumed to be directly behind the blanket. In order to optimize the blanket performance, it was of interest to investigate the effect of varying the reflector composition on the ^{233}U production. The results of this analysis are shown in Table 4-5, where reflectors of all SS or all water as well as a shield of SS and borated water are compared to the reference reflector. It can be seen that the reflector composition has a very small ($<$ 2%) effect on the ^{233}U production at beginning-of-life (BOL). This occurs because the blanket is sufficiently thick to absorb most of the neutrons, at least at BOL. However, the reflector would have a greater effect on the blanket performance after exposure when the neutron population has increased and the leakage is large at the back of the blanket.

4.1.2 TRITIUM BREEDING

In view of the initial operating goals of the DTHR, tritium breeding in the initial blanket concepts was not attempted. As such, the ThO_2 blanket modules were assumed to be placed on the outer portion of the torus only, with shield regions on the inner portion and between the blanket and the toroidal field coils. During the course of operation, a significant number of neutrons are absorbed in both of these shield regions, with a primary source being reflection and back-scattering of neutrons from the outer blanket toward the inner shield region. Later phases of DTHR testing would include tritium breeding blanket modules.

In a commercial hybrid application there is the possibility of placing lithium bearing materials in the front portions of the shield regions and using the available neutrons to breed tritium. Neutronic calculations show that this has

TABLE 4-5

EFFECT OF THE REFLECTOR COMPOSITION

ON BOL BLANKET PERFORMANCE

Reflector Thickness is 15 cm.

REFLECTOR COMPOSITION	RELATIVE ^{233}U PRODUCTION
65% SS 35% H_2O	1.0000
100% SS	1.0157
100% H_2O	1.0093
35% BORATED WATER 65% SS SHIELD	0.9973

the potential for providing from $\sim$ 60% to 80% of the necessary tritium with the boiling water cooled blanket. The remaining tritium could be produced by making the blanket zone thinner and allowing more neutrons to leak into the shields (in the back of the blanket) or by placing tritium breeding blankets above and below the neutral beam injectors and the bundle divertor (in lieu of fissile breeding blankets).

Tritium self-sufficiency could also be enhanced by using some lithium bearing materials in the ThO_2 blanket itself instead of using two separate regions for the Th and Li. This mixing of the Th and Li could have several benefits from a neutronics standpoint, including suppressing the thermal flux and, thus, the fissile burn-out and also lowering the blanket Th inventory, possibly increasing the fissile enrichment. However, it would also decrease the fissile production by allowing neutrons to be absorbed in the Li or lithium-bearing compounds. This would also occur if the blanket thickness were decreased to increase the back leakage. The use of a mixture of Li and Th also offers the possibility of smoothing the increase in blanket power production with exposure by preventing much of the thermal neutron absorption in fissile material. The use of Li compounds in a steam or water cooled blanket presents potential problems with regard to tritium contamination of the water. The use of a mixture of Li and Th materials offers possible advantages in the blanket design for tritium self-sufficiency and will be considered in future work.

4.2 THERMAL-HYDRAULIC ANALYSES

The blanket thermal-hydraulic analyses included the calculation of the peak temperatures and temperature distribution in the hot rod, the DNB ratio (departure from nucleate boiling), and the coolant pumping power. These analyses are discussed in this section.

4.2.1 EXIT COOLANT QUALITIES AND FILM TEMPERATURE DROPS

The blanket operating conditions are such that at the exit of the blanket, forced convective vaporization is the dominant mode of heat transfer. Therefore, at the exit of the blanket, the local coolant quality has a significant effect on the heat transfer coefficient[18] and hence the film temperature drop and the clad temperatures. It can be shown[18] that the two phase heat transfer coefficient for forced convective boiling or vaporization increases with decreasing quality. Hence, the lowest heat transfer coefficient is found at the blanket exit. For radial coolant inflow, the hot rod is also found at the exit end of the blanket. Thus for this flow scheme, hot rod cooling requirements determine the coolant operating conditions allowed. The hot rod film temperature drop was calculated for a hot rod power density of 100 W/cm^3 over a range of exit fluid qualities and for two pressure levels. The results are presented in Figure 4-3. It is clear that the film temperature drop increases gradually with increasing quality up to a quality of $\sim$ 80%. Beyond this quality, the film temperature drop increases rapidly with increasing quality.

On the basis of maximum calculated clad temperature of 265°C and a calculated clad temperature drop of 15°C (at end of life), the maximum clad surface temperature calculated is 250°C. The saturation temperatures corresponding to 26.7 and 33.3 bars are 229°C and 242°C, respectively. The maximum allowed film temperature drops are therefore 21°C and 13°C, respectively. From Figure 4-3 it can be seen that the maximum allowed exit quality is on the order of 90% for a neutron exposure of 1.2 $MW\text{-}Y4/m^2$ (3 year exposure in DTHR). An exit quality of 0.8 was selected for the DTHR to allow for more flexible blanket operations.

4.2.2 FUEL ROD TEMPERATURES

For the boiling heat transfer regime encountered, the two-phase heat transfer coefficient increases with decreasing quality[18], while the fuel rod power density decreases with increasing radial distance from the hot rod. Thus, the fuel rod surface temperatures decrease rapidly, radially away from the hot rod. These effects are illustrated in Figure 4-4 for 1.2 $MW\text{-}Yr/m^2$ of neutron exposure. The calculated hot rod fuel centerline temperature at 1.2 $MW\text{-}Yr/m^2$ neutron exposure is 655°C (1211°F). The temperature distrubution through the hot rod is tabulated in Table 4-6. The reference blanket operating parameters are summarized in Table 4-7.

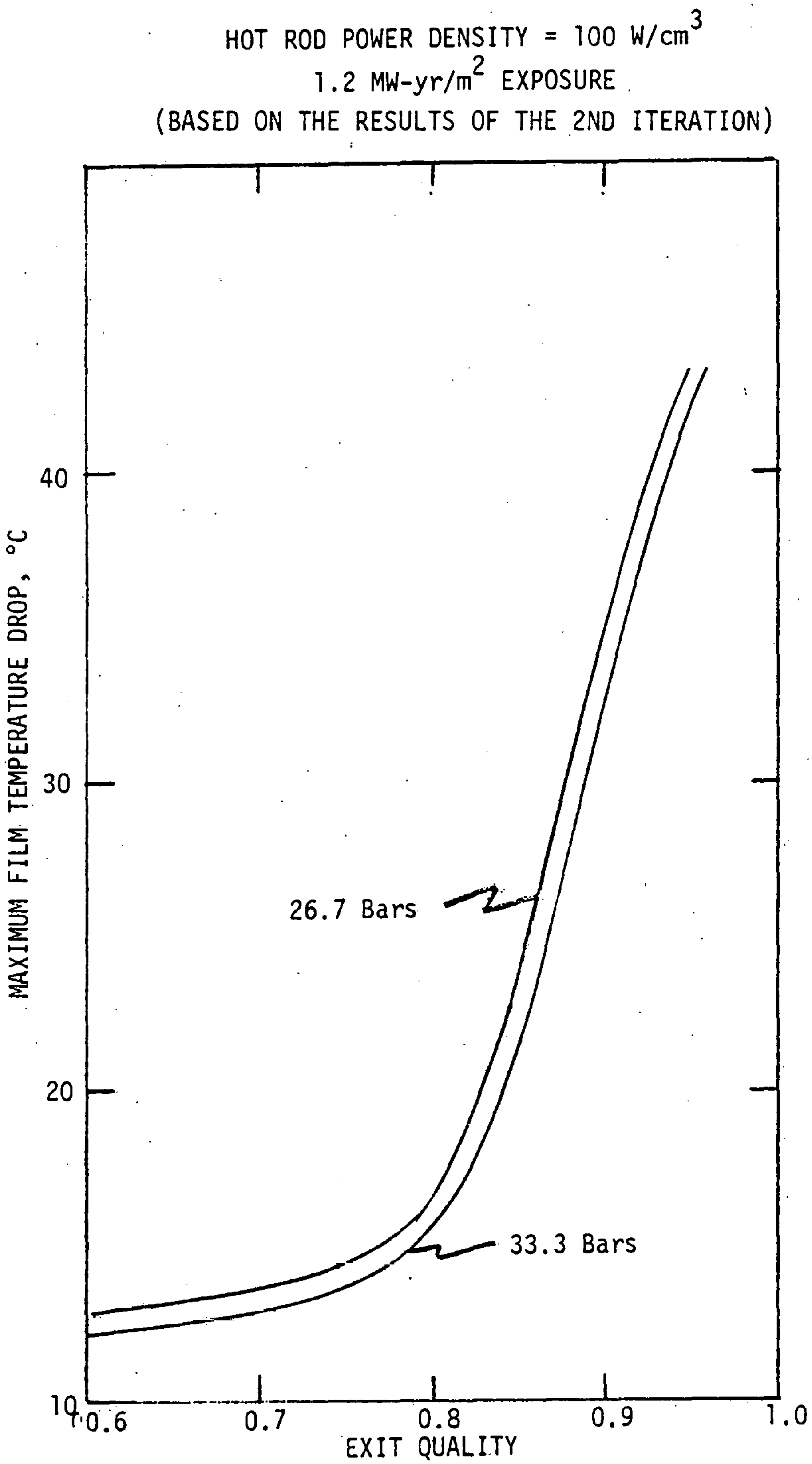


Figure 4-3. Effect of Blanket Exit Steam Quality on Hot Rod Film Temperature Drop.

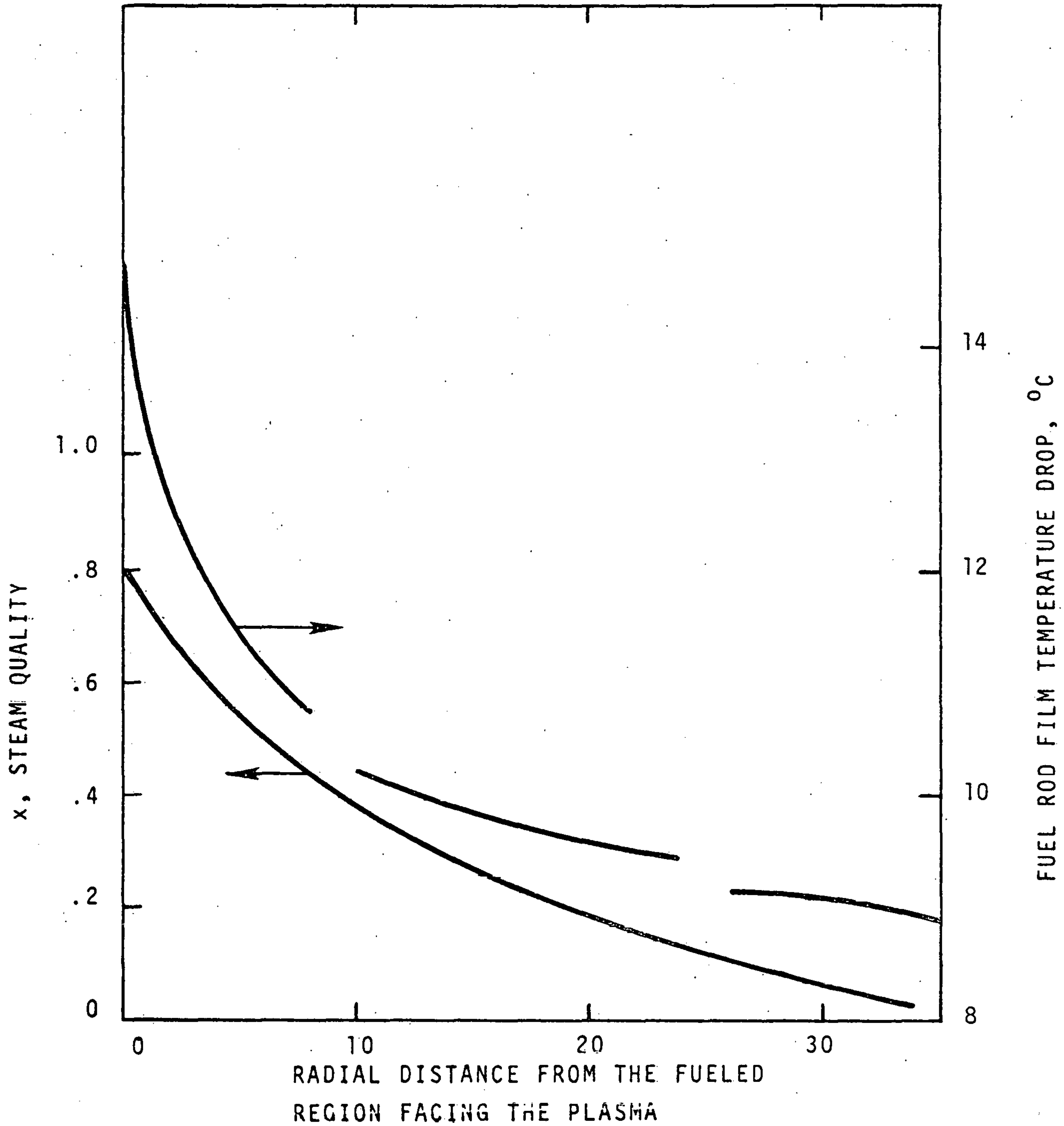


Figure 4-4 Radial Fuel Rod Film Temperature Drop and Steam Quality Distributions

TABLE 4-6

DTHR HOT ROD TEMPERATURE DISTRIBUTION

AT 1.2 MW-yr/m^2 NEUTRON EXPOSURE

PEAK HOT ROD POWER DENSITY, W/cm^3		100
COOLANT OUTLET TEMPERATURE	°C	237
FILM TEMPERATURE DROP	°C	13
CLAD SURFACE TEMPERATURE	°C	250
CLAD TEMPERATURE DROP	°C	15
MAXIMUM CLAD TEMPERATURE	°C	265
HELIUM GAP ΔT,	°C	126
FUEL SURFACE TEMPERATURE	°C	391
RADIAL TEMPERATURE DROP THROUGH ThO_2	°C	264
FUEL CENTERLINE TEMPERATURE	°C	655

TABLE 4-7

REFERENCE DTHR BLANKET OPERATING CONDITIONS

AT 1.2 MW-Yr/m^2 NEUTRON EXPOSURE

AVERAGE BLANKET POWER DENSITY, W/cm^3	15.0
PEAK BLANKET POWER DENSITY, W/cm^3	55
PEAK FUEL ROD POWER DENSITY, W/cm^3	100
MAXIMUM HOT ROD LINEAR POWER, W/cm	165
MAXIMUM WATER MASS VELOCITY, g/cm^2-s	6
INLET COOLANT PRESSURE, BARS	33.3
INLET COOLANT TEMPERATURE, °C	222
OUTLET COOLANT TEMPERATURE, °C	237
INLET WATER SUBCOOLING, °C	20
HOT ROD MAXIMUM HEAT FLUX, W/cm^2	36.25
HOT ROD CLAD SURFACE TEMPERATURE, °C	250
HOT ROD MAXIMUM CLAD TEMPERATURE, °C	265
HOT ROD MAXIMUM FUEL TEMPERATURE, °C	655
DNB RATIO	3.2
COOLANT PRESSURE DROP THROUGH THE BLANKET, BARS	2.4
COOLANT PUMPING POWER TO BLANKET THERMAL POWER, %	0.07

4.2.3 BLANKET MODULE FRONT WALL COOLING

The objective of this analysis was to determine whether the coolant operating conditions can keep the temperature of the blanket front wall below the maximum allowable temperature of the structural material. A schematic of the module cooling system is shown in Figure 3-2.

The preliminary thermal hydraulic analysis of the high quality steam flowing past the module nose indicated that the flow velocities and heat transfer coefficients were not adequate to limit the maximum temperature of the structure to the design limit for the material. Thus, it was necessary to adopt a separate front wall cooling scheme. The flow arrangement suggested is single phase, superheated steam flowing axially parallel to the split cylindrical nose of the module. Superheated (dry) steam was selected to minimize the water number density and excessive high energy neutron moderation. This steam is obtained by flashing the water entering the front wall cooling chamber. The flow path is prescribed by the front wall boundary and a semi-circular flow separator which separates the front wall cooling stream from the exit coolant plenum of the blanket. The operating pressure of the two coolant streams on either side of the flow separator will be approximately equal; the temperature difference between the two streams will be low and equal to the magnitude of superheating allowed in the front wall coolant stream. The use of superheated steam as the front wall coolant has a number of design advantages associated with this choice. A primary factor is that a common coolant allows the design to tolerate leakage across the separator boundary without fear of coolant contamination. Also, the use of a common coolant allows the use of common inlet and exit headers. The front wall coolant can be bled from the primary inlet manifold. At the collecting or exiting manifold, both coolants can utilize the same manifold and piping. The use of a low density superheated coolant for the front wall yields an axially uniform distribution of low water number densities to minimize neutron moderation.

However, the use of a flow separator does increase the structure to fuel volume ratio, degrading the neutronic performance, and increasing the complexity of the mechanical design and the fabrication cost of the module.

Table 4-8 lists the various operating parameters of the front wall cooling system.

4.3 STRUCTURAL ANALYSES

To fully assess the structural integrity of the DTHR blanket module and fuel rod cladding, it was necessary to evaluate these components in relation to criteria that protect against failure over replacement schedules planned during the reactor operation. The structural analysis covered primarily the blanket side walls and the fuel rod cladding. The other components, such as the nose region, rod bundle grids, end caps, inlet and outlet headers, and associated piping were not found to be design limiting.

The structural evaluation of the DTHR duct sidewalls and fuel rod cladding is applicable to a reactor operating with a 40% plant availability over a three year period or 0.6 years of continuous plant operation, i.e. replacement following 1.2 $MW\text{-}Yr/m^2$ of neutron exposure. Over 0.6 years of continuous reactor operation, there is a total of 236,000 plasma on-off cycles. The EOL fast fusion fluence ($E > 0.1$ MeV) is 1.36×10^{22} n/cm^2, while the nuclear heating in the duct walls and the peak power density in the fuel rod are 13 and 100 $watts/cm^3$, respectively.

The DTHR blanket and fuel rod clad stress analysis derived the stresses over the worst case duty cycle and loading. The pressure and thermal stresses were computed from simple linear elastic methods, which are valid, provided that the equivalent stress is less than the associated proportional elastic limit (PEL) stress of the material[11]. Pellet-clad interaction during rapid plasma on-off cycling was not considered because of the available clad ductility at the relatively low EOL fluence in DTHR operation.

The structural evaluation of the DTHR blanket duct side wall and fuel rod cladding to protect against coolant and fuel leakage was based on hypothetical surface cracks at BOL with a depth which slowly grows through the side wall causing an eventual leakage at EOL. Linear Elastic Fracture Mechanics (LEFM) Methods were used to estimate crack growth. LEFM Methods are considered acceptable for the DTHR blanket duct side walls and fuel rod cladding as the

TABLE 4-8

SUMMARY OF OPERATING PARAMETERS
OF THE FRONT WALL COOLING SYSTEM

VOLUMETRIC HEATING RATE, W/cm^3	13
THERMAL ENERGY DEPOSITED IN FIRST WALL AND FLOW SEPARATOR, MW	0.11
COOLANT	STEAM
STATE OF INLET COOLANT	DRY SATURATED STEAM
INLET COOLANT TEMPERATURE, °C	242°
OUTLET COOLANT TEMPERATURE, °C	249°
INLET PRESSURE, BARS	33.3
PRESSURE DROP, BARS	0.1
MASS FLOW RATE, kg/s	4.9
COOLANT FLOW VELOCITY, m/sec	26.8
FLOW CHANNEL CROSS SECTIONAL AREA, cm^2	105.2
FILM COEFFICIENT, W/m^2 °C	1.7
Δ T DUE TO FILM COEFFICIENT, °C	35
Δ T THROUGH FIRST WALL, °C	10
MAX FIRST WALL TEMPERATURE, °C	294
PUMPING POWER, % OF BLANKET THERMAL POWER	0.1

stresses were maintained below PEL. The DTHR blanket duct criteria are similar to the criteria developed for the highly irradiated ORNL blanket module[12], except that coolant leakage by brittle fracture was neglected. The latter was justified as the EOL plane strain fracture toughness in the 20% CW-316-SS DTHR blanket duct wall is expected to be high because of the relatively low neutron irradiation prior to planned replacement. Criteria to protect against excessive deformation failure modes causing hot spots represent a degree of sophistication which was not considered justified for this preliminary concept selection effort.

Based on the available materials data for the 20% cold-worked 316 SS duct and zircaloy clad, the structural evaluation showed that the DTHR blanket module wall and fuel rod clad designs are acceptable for a three year (conservative) replacement schedule (with 1.2 MW-Yr/m^2 of neutron exposure). However, it should be noted that available creep and fatigue crack growth data in a fast fusion neutron spectra are currently not available. The latter data are required to provide confidence in the current design, or prior to extending the time between planned DTHR blanket replacements. A summary of the DTHR blanket module wall and fuel rod clad margins of safety is given in Table 4-9.

It should be noted that oxidation and corrosion effects were not specifically accounted for in this analysis. Since the zircaloy clad is generally considered a viable material for boiling water applications and the stainless steel module walls are relatively thick, corrosion should not be a problem in the short-term DTHR blanket testing. However, commercial applications which intend to reuse the module housings for several refuelings or use higher coolant temperature conditions on the clad will have to be evaluated in detail.

4.4 POWER CONVERSION CONSIDERATIONS

Since one of the major goals of blanket design is to develop a concept that has the capability of appreciable electrical power production, the effect of power conversion efficiency on blanket operating conditions was analyzed. A simple heat transport and power conversion system was assumed. It is a direct conversion steam turbine generator, shown schematically in Figure 4-5. The carryover of radioactive contamiments was assumed to be no worse than that

TABLE 4-9

DTHR BLANKET MODULE MARGINS OF SAFETY

COMPONENT	Allowable CRACK GROWTH (cm)	Calculated CRACK GROWTH (cm)	MS*
DUCT WALL	2.5×10^{-2}	1.4×10^{-2}	0.8
FUEL ROD CLAD	1.6×10^{-3}	1.94×10^{-5}	81.0

$$*MS = \frac{\text{ALLOWABLE VALUE}}{\text{CALCULATED VALUE}} - 1$$

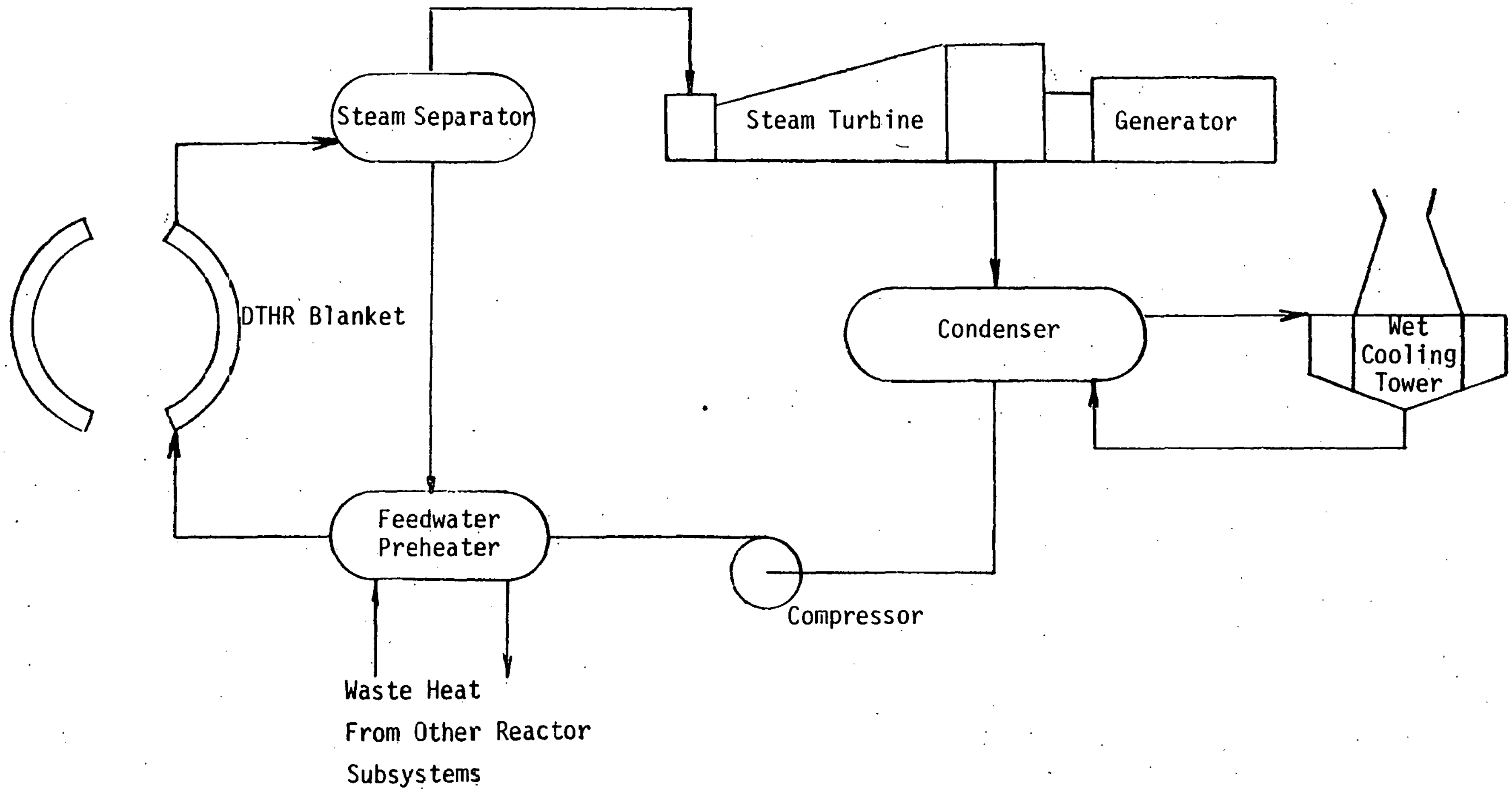


Figure 4-5. Simplified Flow Diagram of DTHR Power Conversion System.

in a BWR. The wet steam produced in the blanket goes to a steam separator and then directly to the steam turbine. The turbine exhaust is condensed in a condenser. The condensate is then heated by waste heat from other subsystems of the DTHR such as the divertor and the inner shield. The feed water then enters the blanket. Thermal storage and energy pulse smoothing requirements are less severe in a commercial reactor because of equilibrium operation and the relatively long duty cycle. These functions could be accomplished by proper design of the steam separator. For the DTHR, the use of an accumulator and a by-pass system would probably be the simplest approach. A direct steam cycle is possible if the tritium accumulation in the blanket coolant as a result of tritium diffusing from the plasma, is sufficiently low. This is achieved in the DTHR by removing the bulk of the tritium (coming from the plasma and diffusing through the first wall) in the separate coolant circuit of the vacuum vessel.

The Steam Rankine Cycle is Carnot - efficiency limited. This implies that the thermal efficiency is dependent on the maximum turbine inlet temperature. This is in turn a function of the maximum allowable fuel clad and blanket structure temperatures. A steam power plant ideal thermal efficiency, η is a product of the efficiency of the Carnot cycle (η_c) the efficiency of the prime mover (η_τ) and the fluid efficiency factor (η_f):

$$\eta = \eta_\tau \eta_c \eta_f$$

The Carnot efficiency is simply related to the turbine inlet steam temperature (T_1) and the condenser temperature (T_2) by:

$$\eta_c = 1 - \frac{T_2}{T_1}$$

The fluid efficiency factor is related to these temperatures and a work factor f such that:

$$\eta_f = \frac{1}{1 + \left(\frac{1}{f} - 1\right) \frac{T_2}{T_1}}$$

Where f is the ratio of energy available as work in the cycle to the energy available in the Carnot cycle. Mackay[13] calculated parametrically the effects of T_2/T_1 and the work factor on the ideal thermal efficiency. In the range of operating conditions of interest, the ideal thermal efficiency can be approximated by the equation:

$$\eta = 0.7874 \left(1 - \frac{T_2}{T_1}\right)$$

A comparison of this equation (with T_2 at 320 K) with the performance of typical water reactor steam power conversion systems is shown in Figure 4-6. It is seen that with the exception of the EBWR (Experimental Boiling Water Reactor), the empirical equation is in good agreement with the gross thermal efficiencies of typical steam power plants. For this reason, the above equation was used for extrapolation to DTHR blanket and steam operating conditions. The ideal thermal efficiency was calculated for a range of blanket operating pressures and for a condenser pressure of 3 in. of mercury as shown in Figure 4-7. It is clear that the thermal efficiency is increased by increasing the blanket pressure. However, an increase in the blanket pressure would result in increased structural materials in the blanket, increased steam densities and increased steam temperatures. The first two are detrimental to neutronic performance in terms of lower fissile breeding; however, thermal power generation can be increased. The last affects the peak clad temperature and limits the thermal power production capability.

Based on considerations of cladding stress corrosion problems, there is an upper temperature-clad life limit on the cladding material. This limit was established at 300°C. Thermal stress considerations on the blanket structure limits the maximum structural wall thickness allowed. It becomes evident that within the constraints imposed by the structural requirements and material temperature limitations, there are design trade-offs which must be investigated in order to arrive at an attractive blanket design suitable for power conversion.

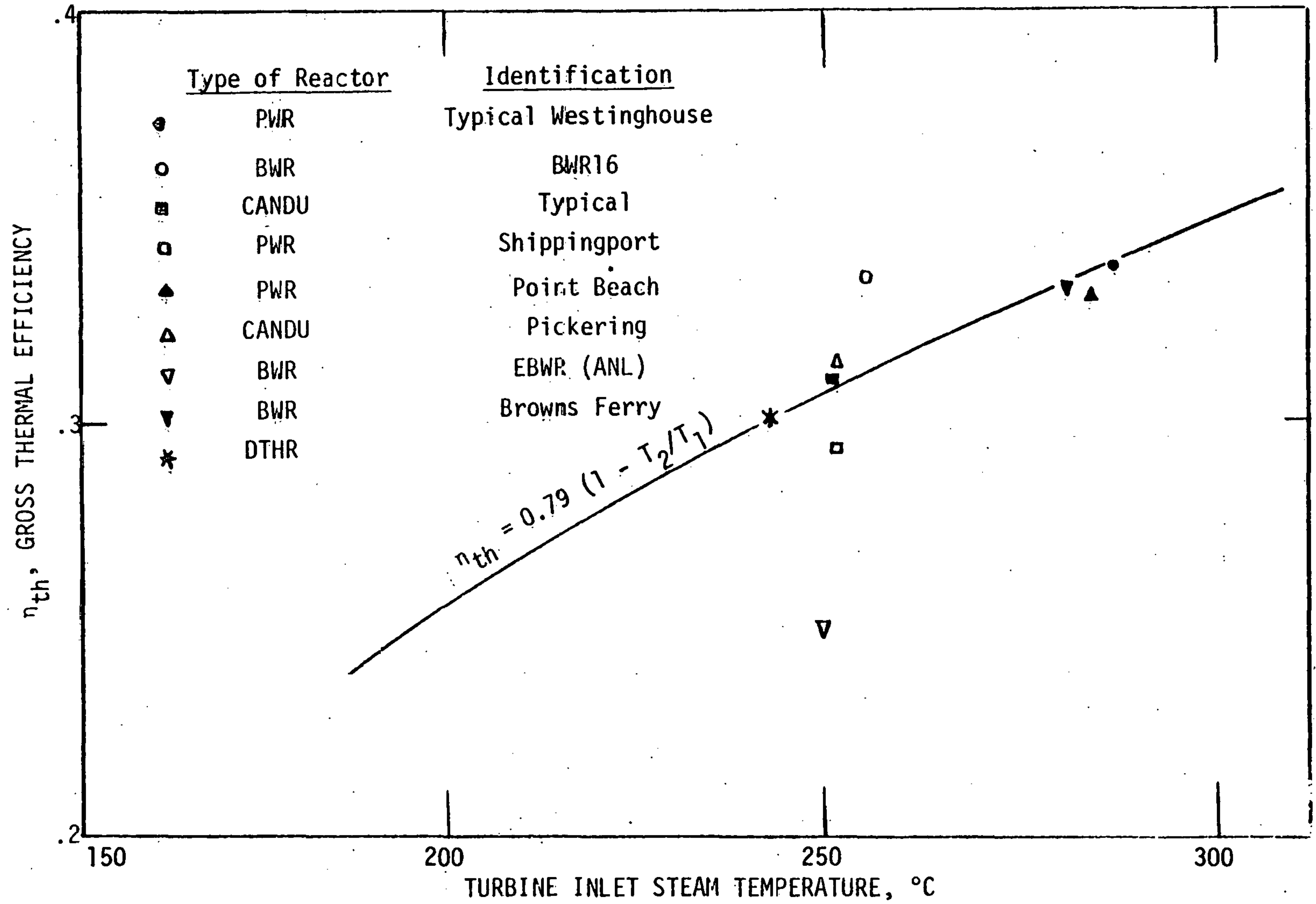


Figure 4-6. Effect of Turbine Inlet Temperature on Gross Thermal Efficiency of Steam Power Conversion Plants. (Condenser Pressure = 3 in Hg)

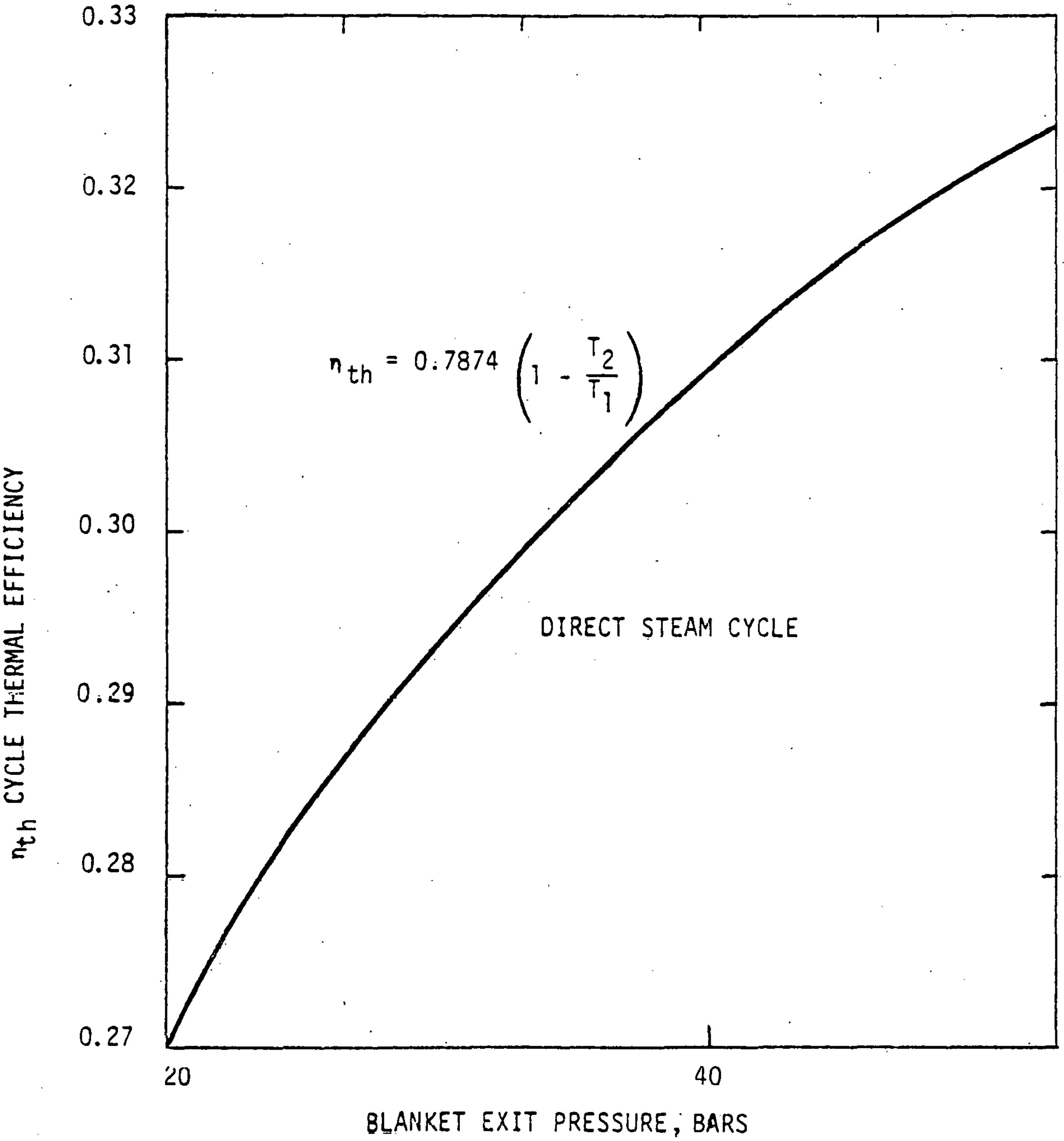


Figure 4-7. Effect of Blanket Pressure on Gross Power Conversion Thermal Efficiency.

4.5 EFFECT OF INCREASED NEUTRON EXPOSURES

4.5.1 NEUTRONIC PERFORMANCE

The performance of a two phase (boiling) water cooled blanket from BOL to an integrated neutron exposure of 8.4 MW-Yrs/m^{-2} is shown in Table 4-10. The blanket composition was changed slightly from that shown in Table 4-2 to reflect differences in structural requirements as a result of differences in thermal stresses over long irradiation times. The fissile production rate decreases by about 27% over this analytical period to a time averaged value of $\sim$ 1200 kg ^{233}U per year and the blanket volume averaged enrichment is $\sim$ 2.2%. Irradiation up to 7 years was analyzed to provide data up to 3% enrichment in the first of the 3 fuel zones (3% is required for LWR fuel). This permits the possible consideration of direct use of the enriched fuel rods without reprocessing. The potential tritium breeding ratio, as measured by the number of neutrons absorbed in the shield regions, increases from $\sim$ 0.6 at BOL to $\simeq$ 0.9 after 8.4 MW-Yr/m^{-2} because of increased ^{233}U fissions. Of this value, $\simeq$ 79% or 0.69 neutrons are absorbed in the inner shield due to a large reflection of neutrons from the blanket towards the inner shield region. The remaining shield absorptions occur from neutrons leaking out the back of the blanket. If tritium self-sufficiency is desired, the blanket could be made thinner to increase the back leakage. Notice also from Table 4-10 that a hypothetical blanket is still far from critical even after seven years of exposure, due to the hard spectrum which results from the specific two-phase water density distribution. If the blanket were flooded as a result of an accident condition, k_{eff} would increase to only 0.87, which is still well below critical.

During exposure, the blanket module will undergo significant neutron radiation damage. A number of important radiation damage parameters in the stainless steel first wall and the zircaloy clad in the boiling water cooled blanket concept are shown as a function of exposure in Figure 4-8. For example, after 8.4 MW-Yr/m^{-2} of irradiation the stainless steel will have $\simeq$ 1290 appm He (maximum) and $\simeq$ 110 DPA with a fluence of $\simeq$ 1.1 x 10^{23} n cm^{-2}. It is uncertain whether the material could withstand this amount of damage, which is also dependent on the operating temperature. Also during this irradiation, the zircaloy clad will have accumulated $\simeq$ 840 appm H from (n,p) reactions in Zr.

TABLE 4-10

BOILING WATER COOLED ThO_2 FERTILE
BLANKET COMMERCIAL PERFORMANCE PROJECTIONS

	AT BOL	AFTER EXPOSURE			
EXPOSURE, $\frac{MW\text{-}Yrs}{m^2}$	—	1.2	2.4	4.8	8.4
F, $\frac{\text{NET NO. } ^{233}\text{U ATOMS}}{\text{FUSION NEUTRON}}$	0.71	0.68	0.65	0.60	0.52
M, ENERGY MULTIPLICATION FACTOR	1.75	3.0	3.8	5.6	8.0
ENRICHMENT, w/o ^{233}U IN Th					
FIRST ZONE (10 cm)	—	0.5	1.0	1.75	3.0
SECOND ZONE (15 cm)	—	—	—	—	2.4
THIRD ZONE (15 cm)	—	—	—	—	1.3
BLANKET AVERAGE	—	—	—	—	2.2
SHIELD ABSORPTIONS, $\frac{\text{NEUTRON}}{\text{FUSION NEUTRON}}$	0.61	0.66	0.69	0.77	0.88
G, $\frac{\text{kg } ^{233}\text{U}}{\text{year}}$ PRODUCTION RATE	1420	1360	1300	1200	1040
CUMMULATIVE ^{233}U, TONNES	—	1.2	2.4	4.9	8.6
AVERAGE ^{233}U PRODUCTION, kg/year	—	1390	1360	1280	1200
k_{eff}, BLANKET EIGENVALUE (SOURCE-OFF)	$\simeq$ 0.02	—	—	—	0.49

4.5.2 EFFECT OF NEUTRON EXPOSURE ON BLANKET OPERATING CONDITIONS

The blanket power increases steadily with neutron exposure because of the increasing build-up of fissile fuel and the corresponding increased fissions. The overall average blanket power density and the blanket peak power density as functions of neutron exposure are given in Figure 4-9 for the reference case. The peak-to-average power density ratio at BOL is 4.78 and drops steadily to a value of 2.6 at 8.4 MW-Yr/m^2 of neutron exposure. However, the peak power density is acutally lower at BOL. The peak to average value of 2.6 is more representative of what might be expected at equilibrium operating conditions. The power density distributions for these levels of neutron exposure are given in Figure 4-10 for the reference case. It is evident that the power density is flattened appreciably with increasing neutron exposure. It should be pointed out that these results do not inlcude any fuel shuffling. The power density distributions at higher neutron exposures are expected to be more representative of those for equilibrium blanket operating conditions with fuel shuffling.

With increasing neutron exposure and blanket power, the clad surface temperature remains relatively constant if the blanket exit steam quality is maintained at 80%. This is due to the fact that the mass flowrate and the heat transfer coefficient increase with increased power production. However, the maximum clad and fuel centerline temperatures increase steadily as shown in Figure 4-11, although the maximum clad temperature increases slowly with neutron exposure. At the end of the 8.4 MW-Yr/m^2 neutron exposure period analyzed, the maximum clad and fuel centerline temperatures reach 325°C and 1778°C, respectively. These temperatures are comparable to those of typical light water reactors. The clad exposure to fast neutrons in a fusion blanket may limit the maximum allowable clad temperature to a lower value. In any case, a steady fuel rod exposure of 8.4 MW-Yr/m^2 is not conceivable even for commercial operations, where fuel shuffling will be used to optimize fuel management as well as power production; thereby reducing the integrated neutron exposure of the fuel rods by a factor of 2 or 3. This level of exposure would be considered for the module structure, however, since it would permit reuse of the pressure housing for 3 or 5 refuelings.

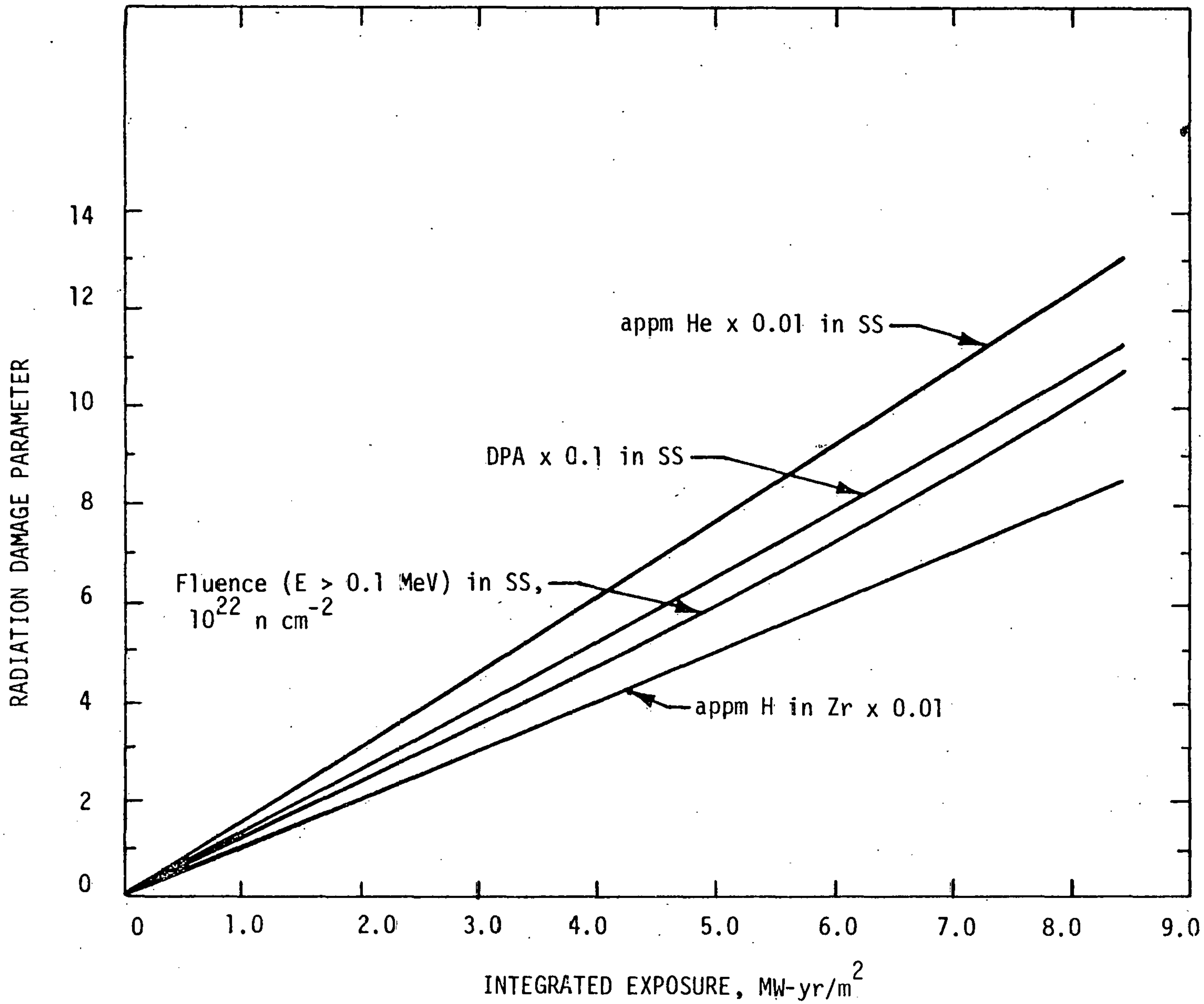


Figure 4-8. Radiation Damage Parameters as a Function of Exposure for the Boiling Water Cooled Blanket

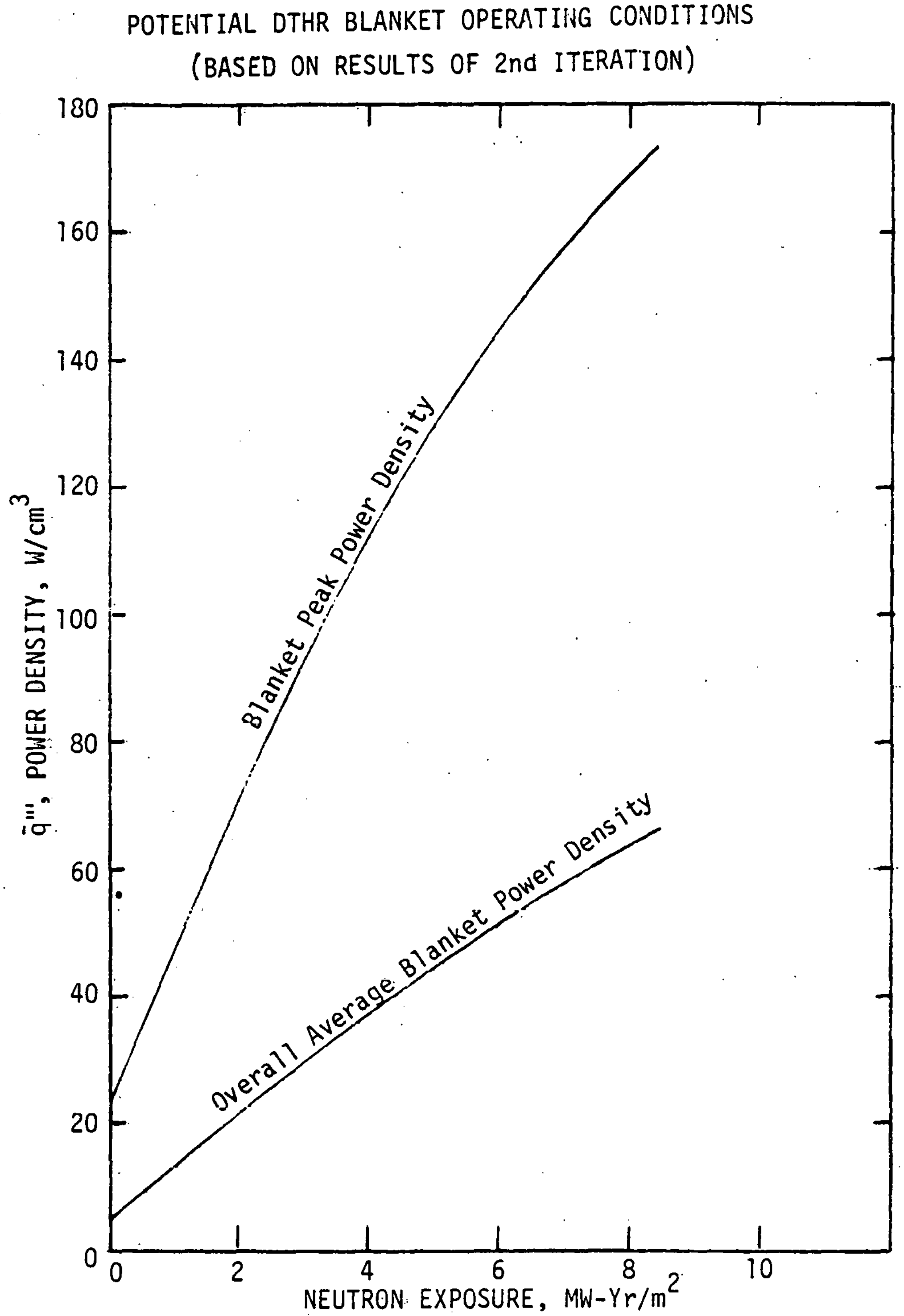


Figure 4-9. Effect of Neutron Exposure on Blanket Power Densities No Fuel Shuffling.

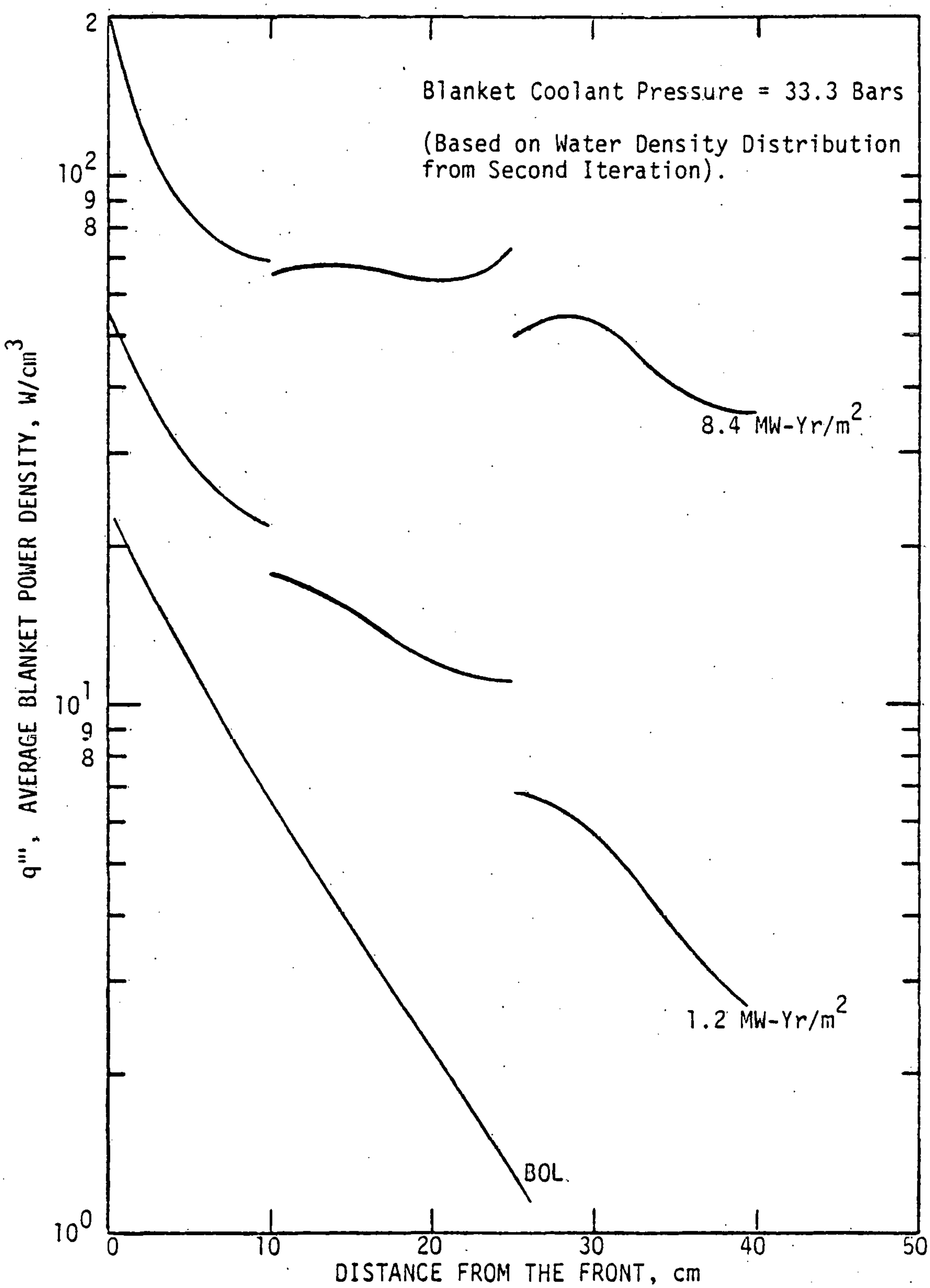


Figure 4-10. Effect of Neutron Exposure on Balnket Power Density Distributions. No Fuel Shuffling.

REFERENCE BLANKET OPERATING CONDITIONS

NO FUEL SHUFFLING

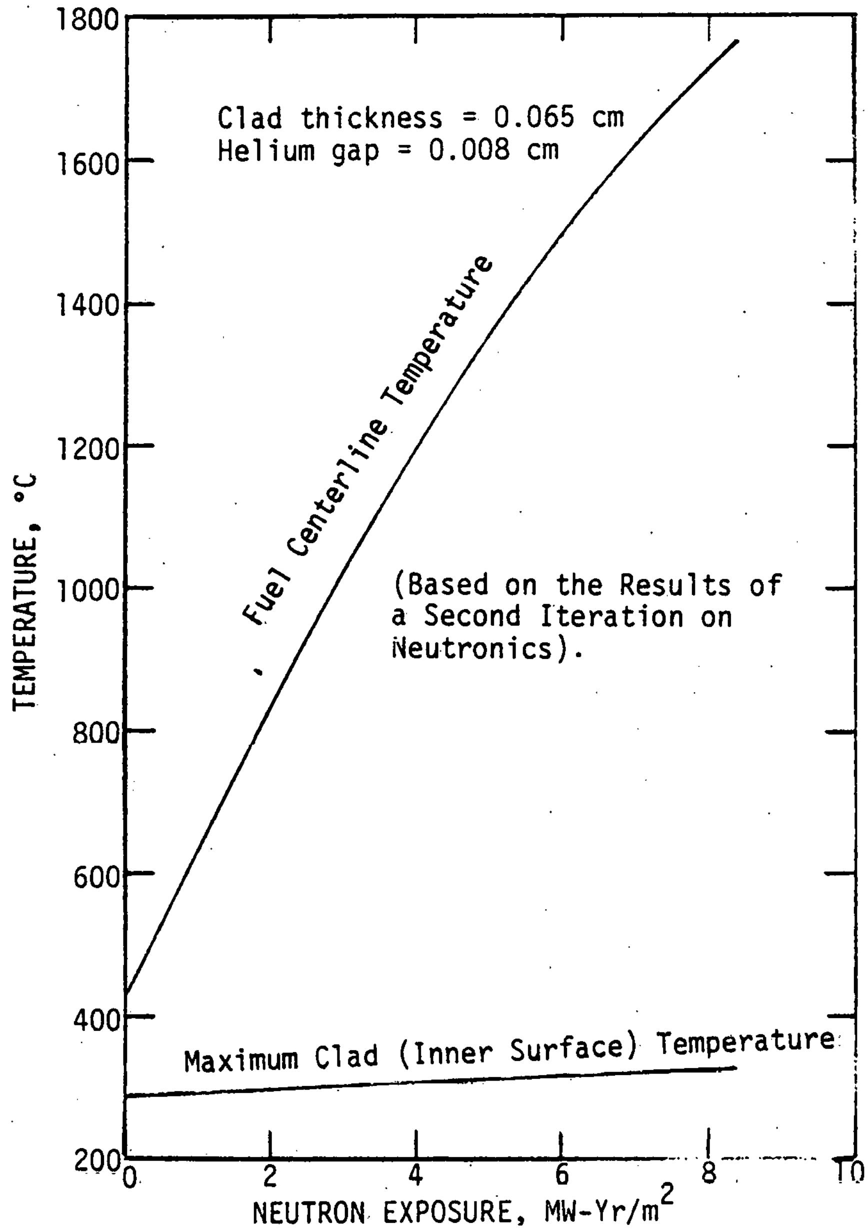


Figure 4-11. Effect of Neutron Exposure on Maximum Fuel and Cladding Temperatures.

4.5.3 CRITICAL HEAT FLUX AND DNB RATIOS

The critical heat flux was estimated based on the Jens and Lottes correlation [14] with the assumption of 33.3 bars of pressure and subcooling such that $\Delta T_{sub}^{0.22} = 1.0$ (at the exit of the blanket), i.e.

$$(q'')_{crit} = 0.817 \times 10^6 \left(\frac{G}{10^6}\right)^{0.16}$$

This correlation was used instead of more recent correlations because the latter are generally applicable to higher pressures and very restricted operating conditions. The mass velocity (G) was based on the nominal value for an exit quality of 0.8. The DNB ratios (ratio of critical heat flux to maximum hot rod heat flux) were then calculated for the hot rod.* The results (given in Figure 4-12) show that the DNB ratio at the beginning of life is 5.3, dropping steadily to a value of 1.2 at 8.4 MW-yr/m^2 of neutron exposure. The maximum hot rod heat flux is 16 W/cm^2 at BOL and increases to 127 W/cm^2 at 8.4 MW-yr/m^2. Thus, at equilibrium blanket operating conditions, typical of commercial reactors, the DNB ratio would be relatively low.

4.5.4 PRESSURE DROP AND COOLANT PUMPING POWER

The total pressure drop through the blanket consists of the friction losses due to boiling two phase flow, acceleration losses, entrance and exit losses, and expansion and contraction losses. The largest contribution to the total pressure drop was found to be from the friction losses. On the basis of fixing the steam exit quality constant at 80%, the pressure drop increases steadily with neutron exposure because of the increased water flow rate required. In order to maintain a constant pressure to the steam turbine, the blanket inlet pressure must be increased with time. This is illustrated in Figure 4-13. The coolant pumping power was calculated by assuming that the total pressure drop through the primary coolant system is twice that computed for the blanket alone. The coolant pumping power required as a percent of the blanket power is given in

† This correlation was assumed for convenience in light of the uncertainty of the applicability of other correlations to cross flows and to the relatively low reference pressure.

* The DNB power ratio was not calculated because this involves detailed consideration of potential over-power mechanisms which are expected to be appreciably different from fission reactors because of the non-critical nature of the blanket. Such considerations are not within the scope of this study.

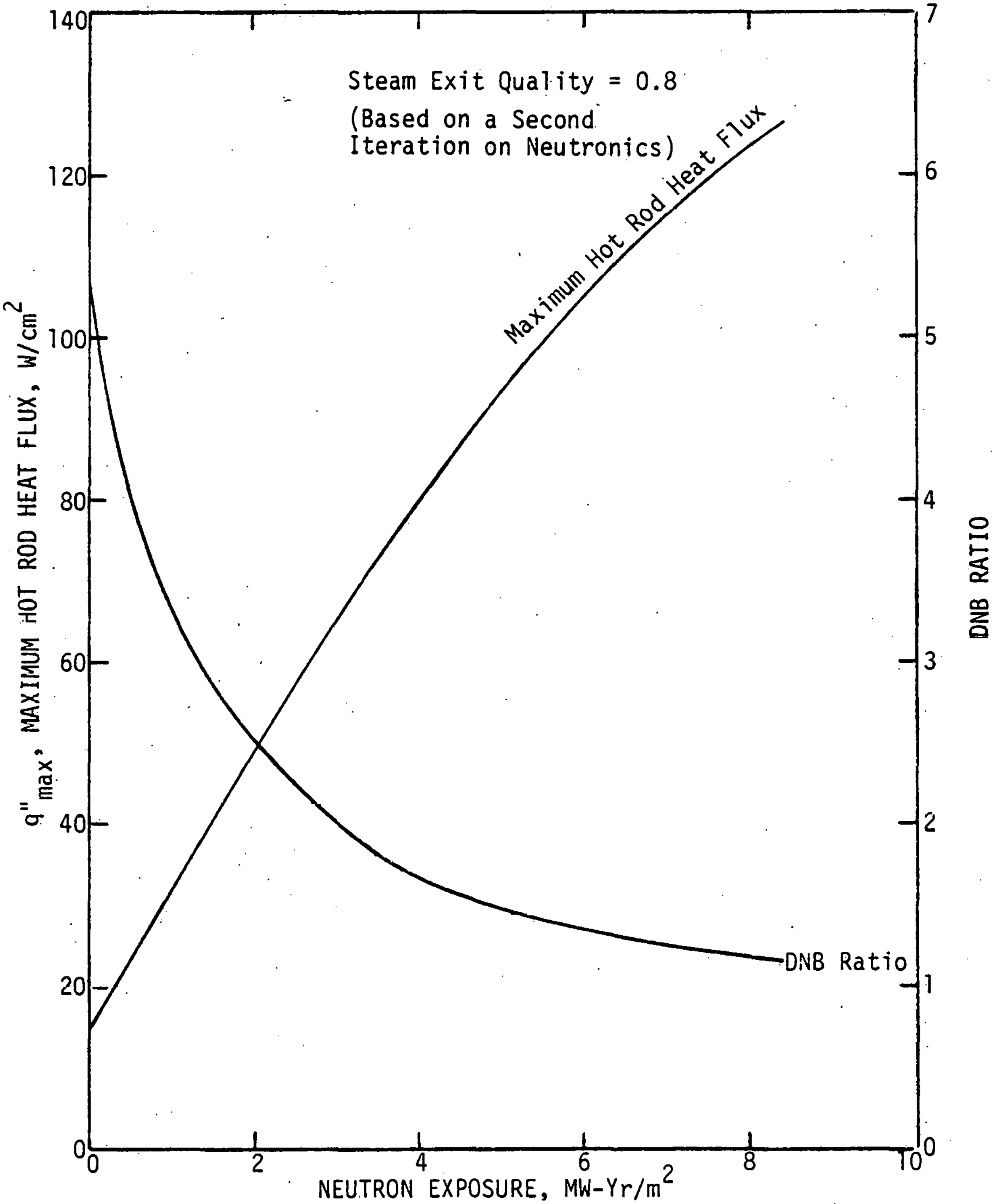


Figure 4-12. Effect of Neutron Exposure on Hot Rod Heat Flux and the DNB Ratio.

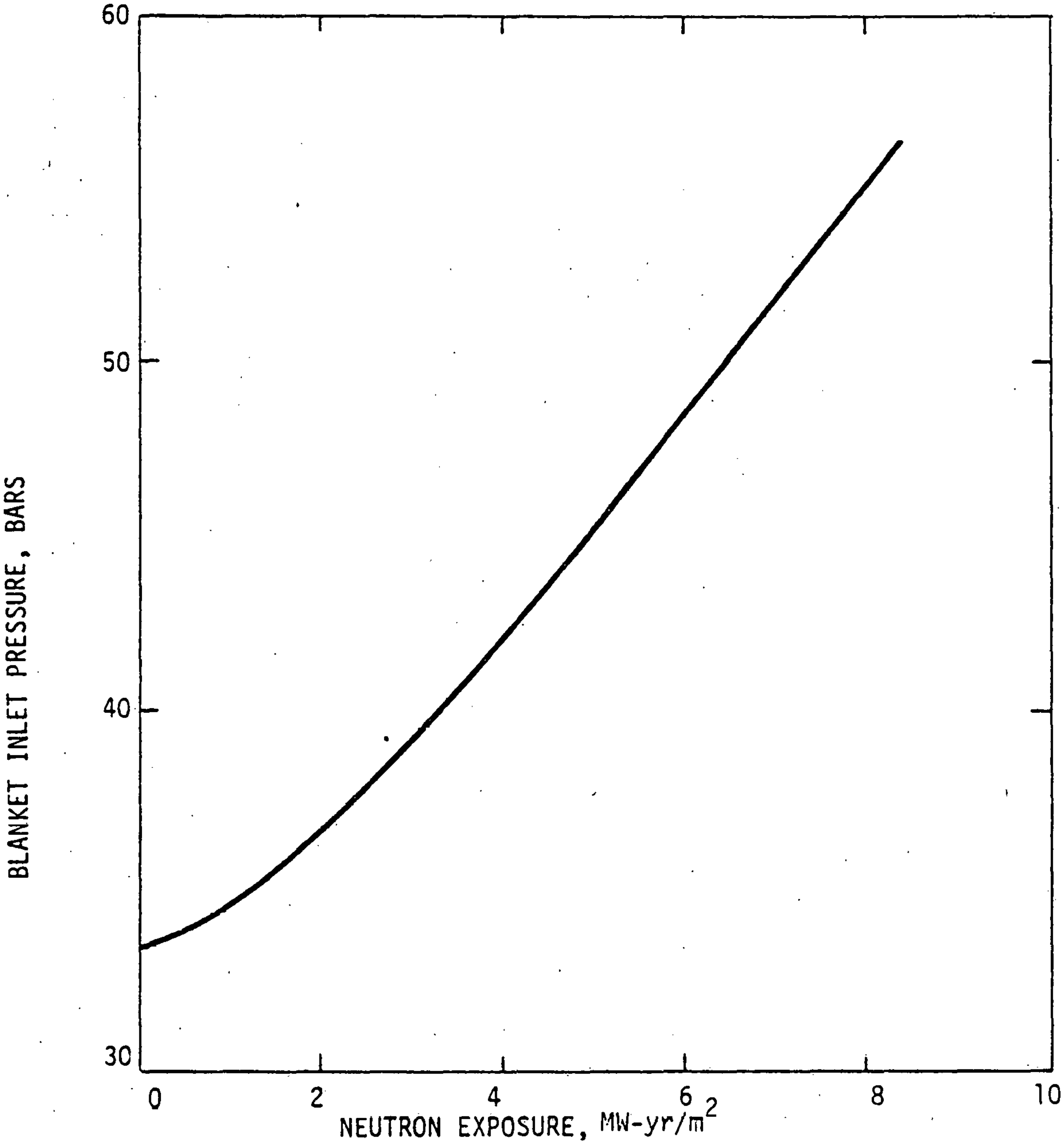


Figure 4-13. Effect of Neutron Exposure on Required Blanket Inlet Pressure.

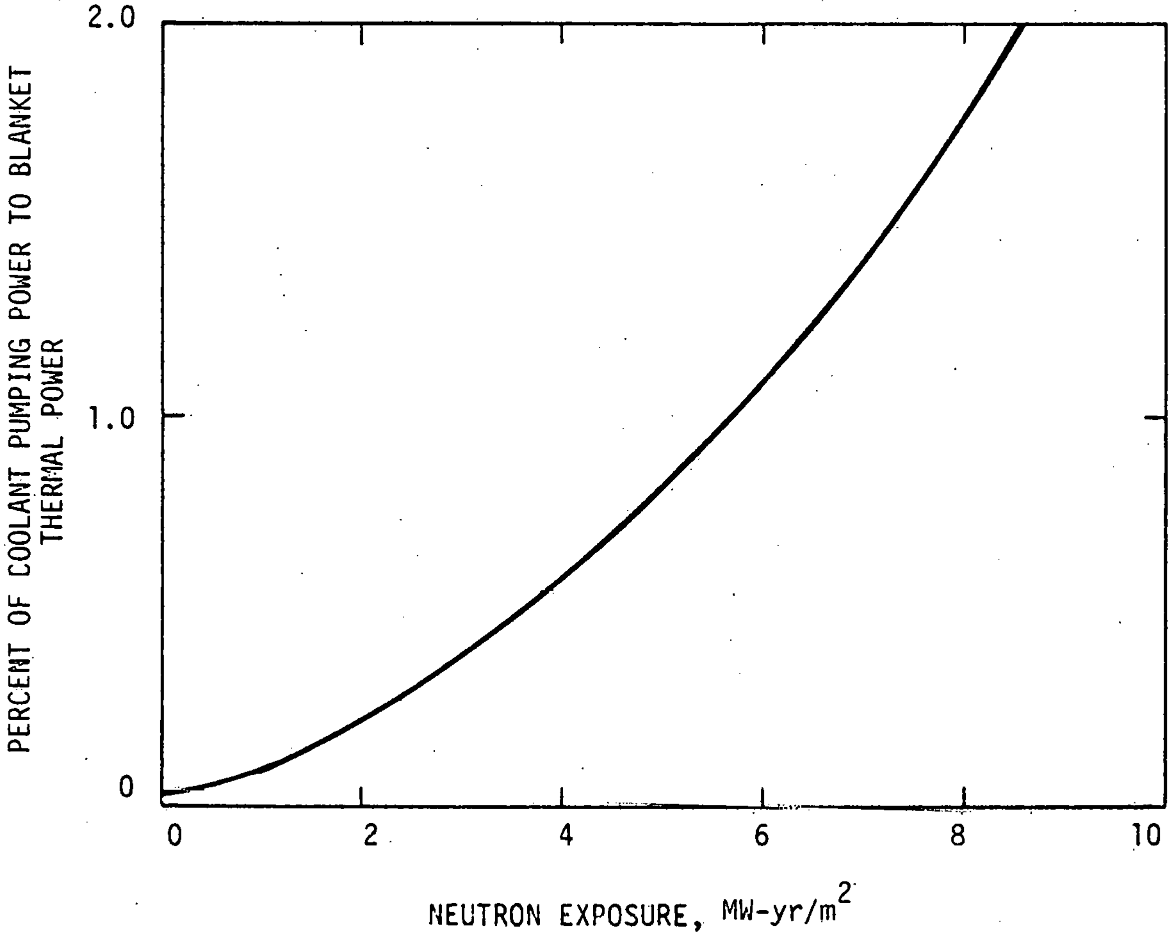


Figure 4-14. Effect of Neutron Exposure on Coolant Pumping Power Requirements for Constant Clad Temperature.

Figure 4-14. It is seen from the latter figure that the coolant pumping power is extremely modest. This suggests that at 8.4 MW-yr/m^2, the required pumping power is only 1.9% of the blanket thermal power. It should be emphasized that these calculations merely explore the operational limits in a DTHR. Actual operations are expected to involve relatively low neutron exposures. The higher exposures do provide an indication of the achievable performance for equilibrium blanket operating conditions.

4.6 NEUTRONIC COMPARISON OF WATER COOLED AND BOILING WATER COOLED BLANKET CONCEPTS

In this section the performance under long irradiation times of a boiling water cooled blanket module is compared to that of the low temperature, low pressure water cooled blanket concept developed in early stages of this program and presented in detail in References 1 and 15. The blankets are considered for use in a commercial reactor with a plant factor averaged wall loading of 1.2 MW m^{-2} and with blanket coverage parameters as discussed in section 4.1 and Table 4-1.

The blanket material compositions for the two-phase water and single-phase water cooled designs are shown in Table 4-11. The two-phase water cooled blanket was made 15 cm thicker than the water cooled concept in order to keep the back blanket neutron leakage about the same at BOL. This was necessary because of the decreased moderation with the two-phase water, and thus results in a higher blanket fuel inventory. Another major difference between the two concepts is in the blanket module structural material. The water cooled blanket was conceived for low temperature and pressure operation, which permitted the use of Zircaloy-4 as the structural material. For the higher pressure two-phase water cooled concept, it was necessary to use stainless steel as the structural material. This will affect the neutronic performance somewhat since the steel has a higher parasitic absorption cross section than the zircaloy. However, with the two-phase water coolant the neutron spectrum will be harder and the parasitic losses will not be as important as in the softer water cooled spectrum. Indeed, the main objective in utilizing steam coolant was to prevent the thermal fissioning of the ^{233}U, which was found to occur [1,15] in the water cooled case, and at the same time to provide for the possibility of electrical power generation by operating at higher coolant temperatures and pressures. The boiling water cooled blanket composition shown in Table 4-11 is slightly different than the one discussed in section 4.1, and represents a somewhat optimized design in which the blanket module wall also serves as the first wall facing the plasma.

In Figure 4-15 the fissile enrichment in the first 10 cm and the net number of fissile atoms per fusion neutron for the two blanket concepts are shown as a function of the integrated exposure. Note that in the commercial application exposure is 1.2 MW-Yr/m^{-2}, so that a value of 8.4 MW-Yr/m^{-2} represents seven years of exposure time. The two-phase water cooled, stainless steel structure blanket reaches an enrichment of $\sim$ 3% in the first 10 cm in a nearly linear rate of increase, while the water cooled, zircaloy structure blanket reaches a near equilibrium enrichment $\sim$ 1.86% after about four MW-Yr/m^{-2}. Correspondingly, the net ^{233}U production per fusion neutron approaches zero in the water cooled case while it is still fairly large in the two-phase water cooled design.

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In comparing these two concepts, it is evident from Figures 4-15 and 4-16 that the water cooled concept has higher short term fissile enrichment and production capabilities relative to the two-phase water cooled case but also undergoes large increases in the blanket multiplication. The reason for this is the increased neutron moderation with the water coolant, which allows the neutrons to be slowed down and captured in a thinner blanket, thereby producing a higher enrichment. However, the increased moderation also results in a large amount of thermal neutron fissions in the ^{233}U, causing the net fissile

TABLE 4-11

COMPARISON OF MATERIAL CONPOSITIONS FOR THE WATER COOLED AND BOILING WATER COOLED BLANKET CONCEPTS

	TWO-PHASE BOILING WATER COOLED	WATER COOLED
BLANKET FUEL SECTION THICKNESS, cm	40	25
STRUCTURAL MATERIAL	STAINLESS STEEL	ZIRCALOY-4
ThO_2 VOLUME FRACTION	0.454	0.569/0.386*
H_2O VOLUME FRACTION	0.306	0.241/0.474*
Zr VOLUME FRACTION	0.095(clad only)	0.176/0.130*(clad plus structure)
SS VOLUME FRACTION	0.135	——
WATER DENSITY, $\frac{gms}{cm^3}$		
FIRST ZONE (10 cm)	0.032	1.00
SECOND ZONE (15 cm)	0.080	1.00
THIRD ZONE (15 cm)	0.513	——

*First Zone/Second Zone (2 zones only with different compositions in the two zones, See Table 3-3)

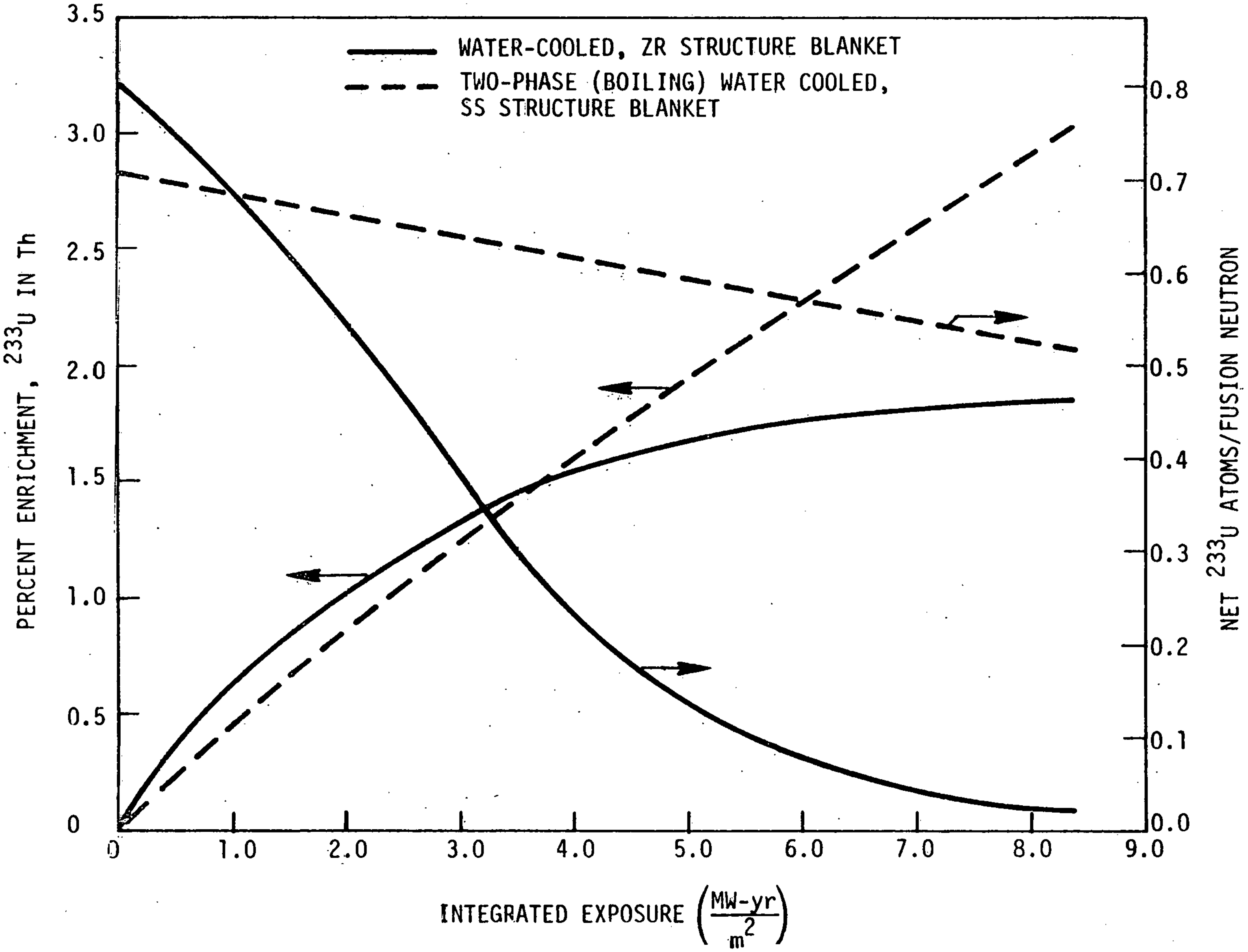


Figure 4-15. Comparison of the Water and Boiling Water Cooled Blanket Enrichment and Fissile Atom Production as a Function of Exposure. (Based on Third Neutronic Iteration).

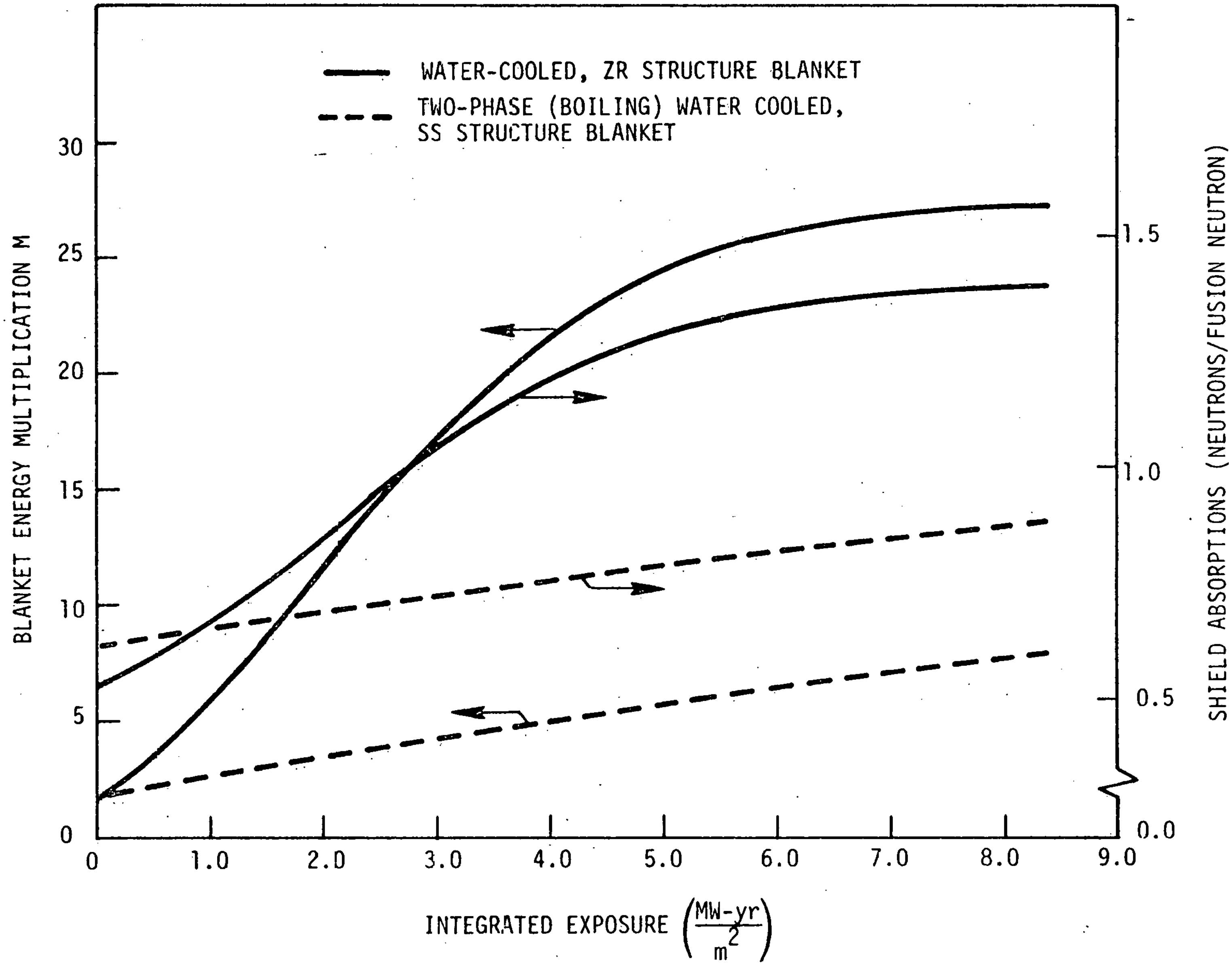


Figure 4-16. Comparison of the Energy Multiplication and Number of Shield Absorptions (≈ Tritium Breeding Ratio) in the Boiling Water and Water Cooled Blankets as a Function of Exposure. (Based on Third Neutronic Iteration).

production to decrease and the blanket energy multiplication to increase fairly rapidly. Another important factor in the performance comparison of the two blankets is the different structural materials. The stainless steel used in the two-phase boiling water cooled blanket has a larger parasitic neutron absorption rate compared to the ^{233}U production rate, causing F to be lower at BOL in the boiling water blanket, as shown in Figure 4-15.

The differences in the neutron flux as a function of energy for the two concepts are shown in Figure 4-17 for two different positions in the blanket. As expected, there are dramatic differences of up to an order of magnitude in the thermal flux and the fast flux for the two coolants. As mentioned previously, the main reason for using two-phase water was to achieve a harder neutron spectrum in the blanket and thereby prevent the ^{233}U thermal neutron fissioning. As Figure 4-17 shows, this has been accomplished with two-phase water as blanket coolant.

Another quantity of interest in comparing the two blanket concepts is the spatial variation of the Th capture and ^{233}U absorption reaction rates in the blanket. These variations are shown in Figure 4-18 for the boiling water cooled blanket and in Figure 4-19 for the water cooled blanket. For the water cooled case the ^{233}U absorption rate is nearly equal to the ^{233}U production rate from the Th (n,γ) reactions after seven years of irradiation, so that the net fissile production is about zero. However, in the boiling water cooled case, the ^{233}U absorption reactions have been suppressed so that the net fissile production stays fairly large. In both blankets the average ^{233}U capture-to-fission ratio (α) is in the range 0.12 to 0.13, which indicates that $\sim$ 88% of the ^{233}U absorption reactions are fission events. The different neutron spectrum and thermal fission rates also result in much different values in the blanket criticality constant for the two concepts. For the boiling water cooled case, $k_{eff} = 0.487$ after seven years of irradiation while for the water cooled case $k_{eff} = 0.87$ after the same length of exposure. In the event of a water flooding accident, the keff for the boiling water cooled blanket after seven years of irradiation remains subcritical (keff = 0.92). This is due to the fact that there is significant leakage from the blanket when compared to a fission reactor.

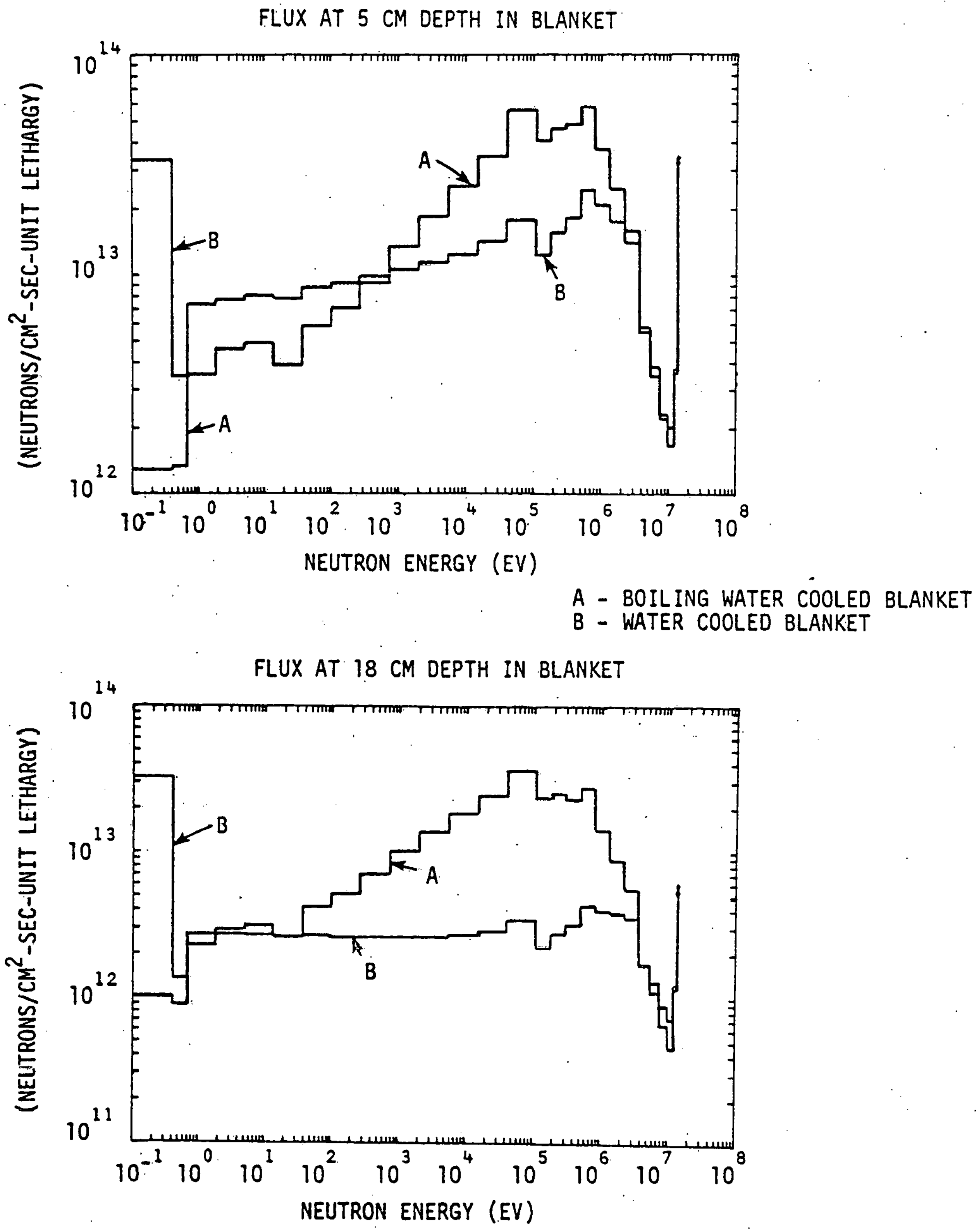


Figure 4-17. Plot of the Neutron Flux as a Function of Energy for the Boiling Water Cooled and Water Cooled Blankets at Two Different Depths in the Blankets.

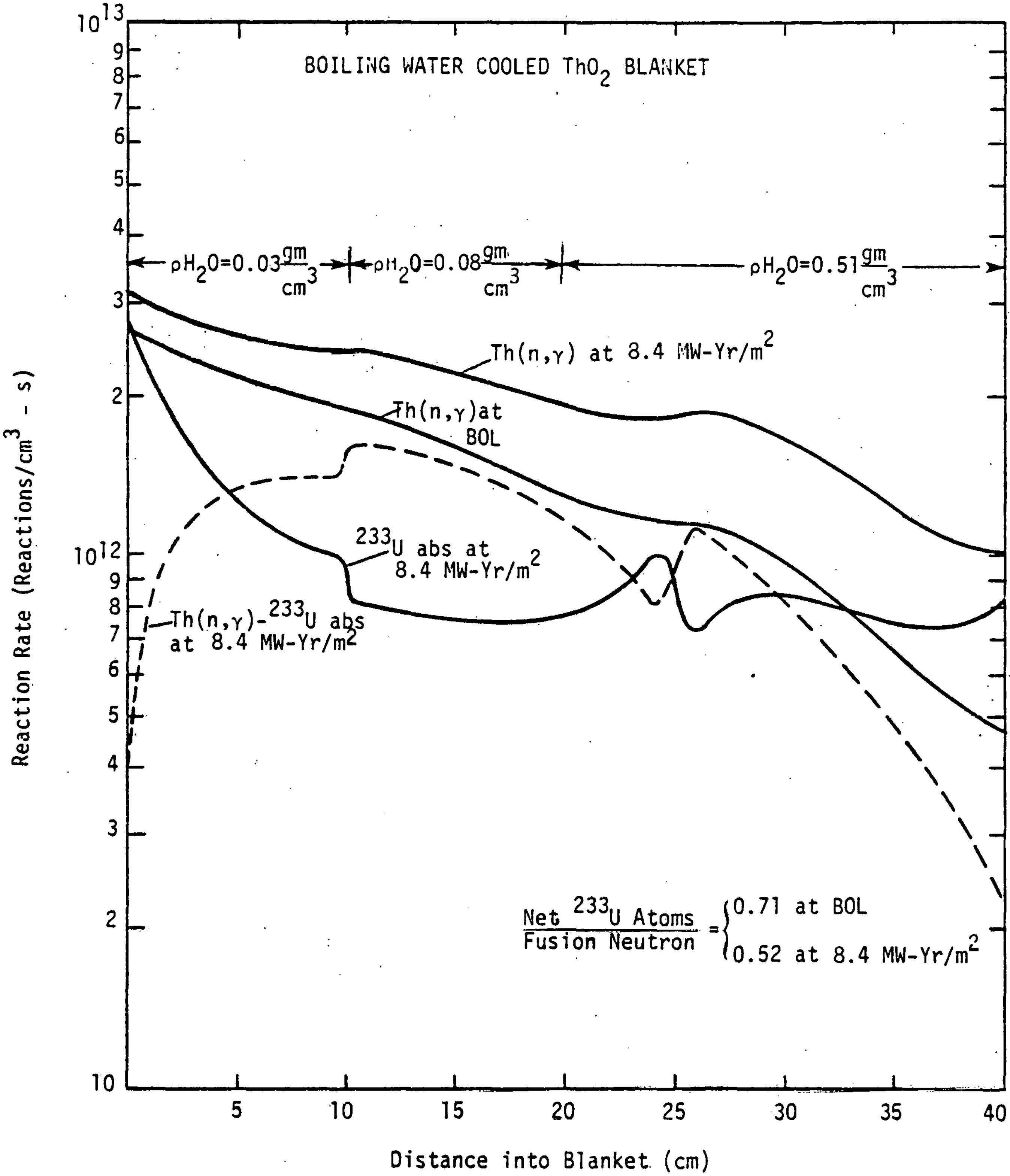


Figure 4-18. Spatial Variation of the Th (n,γ) and ^{233}U Absorption Reaction Rates in the Boiling Water Cooled ThO_2 Blanket.

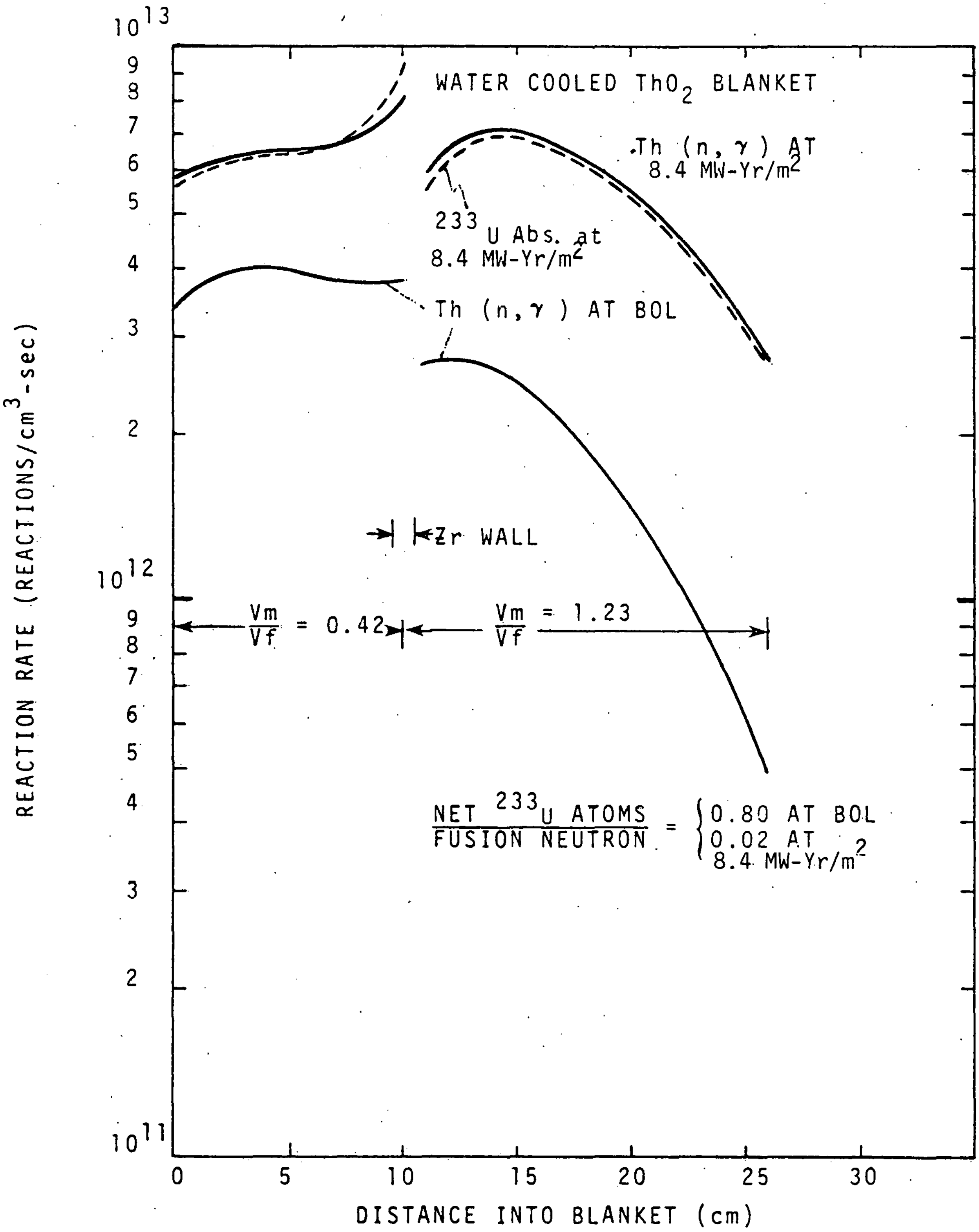


Figure 4-19. Spatial Variation of the Th (n,γ) and ^{233}U Absorption Reaction Rates in the Water Cooled ThO_2 Blanket.

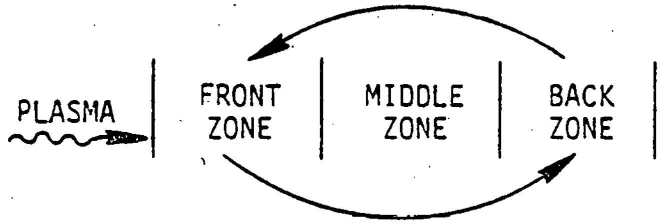
5.0 COMMERCIAL HYBRID REACTOR DESIGN IMPLICATIONS

It should be emphasized that the neutronic calculations performed have been aimed at the design of a DTHR blanket, where the blanket may be tested in the reactor for a period of perhaps up to 3 years. Accordingly, lifetime calculations were carried out without regard to fuel shuffling, optimum fuel management, and equilibrium blanket operating conditions. A study of a commercial hybrid blanket design, operating conditions and performance is not within the scope of this phase of the program. Nevertheless, the neutronic calculations performed to date can provide some insight on the capabilities for fissile and power production in a commercial application and suggest preliminary operating conditions.

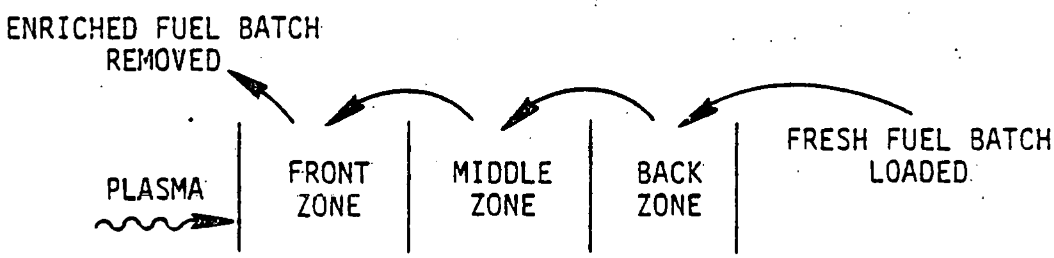
5.1 FISSILE AND POWER PRODUCTION IN A COMMERCIAL HYBRID

The fissile and power production capabilities of a commercial blanket were estimated by assuming a maximum residence time of the fuel rods in the blanket of 7 years (8.4 $MW\text{-}Yr/m^2$) and that fuel shuffling occurs every 2.3 years. The fuel shuffling scenario assumed is that shown in Figure 5-1b. The estimation was based on neutronic calculations for the reference blanket model that included a separate vacuum vessel. The results of this analysis are summarized in Table 5-1. It should be pointed out that the fuel enrichment indicated and the average fissile production rate can be achieved separately, but not together.

For example, the 3.03% ^{233}U concentration in thorium is achieved after 7 years of irradiation and occurs in the first 10 centimeters of the blanket, while the time-averaged ^{233}U production rate of 1224 kg/yr applies to the entire blanket with an average enrichment of 2.2%. These different fuel production rates, along with other quantities, are shown by zones in the blanket in Table 5-2. As the table shows, after 7 years of irradiation, about 24.5% of the fuel is enriched to 3.03%. Indeed, the achievement of a product with an enrichment of this magnitude is of interest primarily from the standpoint of the "no reprocessing" fuel cycle scenarios (see page 5-5).



a) Rotation Scheme, whereby the Front and Back Blanket Zones are Interchanged Periodically while the Middle Zone is Stationary.



b) Inward Shuffling Scheme, whereby Batches of Fuel are Shuffled to Inner Blanket Zones Periodically as Fresh Fuel is Loaded in the Back Zone and Enriched Fuel is Removed from the Front Zone.

Figure 5-1. Two Possible Blanket Fuel Management Schemes.

TABLE 5-1

POTENTIAL COMMERCIAL FISSILE AND POWER PRODUCTION CAPABILITIES

AVERAGE ^{233}U PRODUCTION RATE AT EQUILIBRIUM BLANKET OPERATING CONDITIONS, kg/yr	1220
^{233}U CONCENTRATION IN Th FUEL PRODUCT AFTER SEVEN YEARS RESIDENCE TIME (WITH FUEL SHUFFLING), (8.4 MW-Yr/m^2), %	3.0
DUTY CYCLE AVERAGED THERMAL POWER PRODUCTION AT EQUILIBRIUM BLANKET OPERATING CONDITIONS, MW_t	4000
GROSS PLANT THERMAL EFFICIENCY, %	29
GROSS DUTY CYCLE AVERAGED ELECTRICAL POWER PRODUCTION AT EQUILIBRIUM BLANKET OPERATING CONDITIONS, MWe	1160

TABLE 5-2

COMPARISON OF IMPORTANT BLANKET NEUTRONIC PARAMETERS BY ZONES IN THE BLANKET

	FIRST ZONE	SECOND ZONE	THIRD ZONE
ZONE THICKNESS (cm)	10	15	15
FRACTION OF BLANKET VOLUME	0.245	0.374	0.381
FRACTION OF NET ^{233}U PRODUCTION			
- AT BOL	0.387	0.392	0.221
- AFTER 1 YEAR	0.365	0.402	0.233
- AFTER 7 YEARS	0.289	0.466	0.245
^{233}U ENRICHMENT			
- AFTER 1 YEAR	0.53%	0.36%	0.21%
- AFTER 7 YEARS	3.03%	2.44%	1.34%
FRACTION OF BLANKET POWER			
- AT BOL	0.696	0.255	0.049
- AFTER 1 YEAR	0.556	0.290	0.154
- AFTER 7 YEARS	0.403	0.316	0.281

(> 3% fuel enrichment is the desired goal for this purpose). If fuel reprocessing, fissile separation and concentration were to be considered, then the average ^{233}U production rate of 1220 kg/yr can be achieved.

If it is assumed that blanket power production can be maintained relatively high by fuel shuffling and that equilibrium operating conditions comparable to those of the DTHR at 8.4 MW - yr/m^2 can be obtained, then the commercial performance can be estimated as given in Table 5-1. It must be emphasized that this performance represents an approximation only, because the fissile and power density distributions of an equilibrium cycle blanket are expected to be somewhat different from those assumed. The characteristics of such a commercial hybrid are compared with typical water reactors in Table 5-3. It is clear from these comparisons that all the operating conditions are well within or near those of conventional water reactors. Many of the indicated parameters can be altered, subject to design modifications. In particular, the minimum DNBR is based on the DTHR calculations, i.e., irradiation in the front zone of the 3 zone blanket. With fuel shuffling in the commercial reactor, the maximum power density could be reduced and the minimum DNBR increased for the same overall neutronic performance.

5.2 EFFECT OF FUEL SHUFFLING

While the effect of fuel shuffling on blanket fissile and power production were not analyzed in this DTHR program, the extensive neutronic analyses performed to date permit a qualitative assessment of the trends that may be expected.

5.2.1 FUEL MANAGEMENT ALTERNATIVES

In general, optimum fuel management can be determined only by carrying out comparative economic evaluations of alternate fuel management scenarios that include symbiosis with LWR's and perhaps LMFBR's as well. Three possible fuel management scenarios are illustrated in the flow diagrams of Figure 5-2. Table 5-4 compares some of their respective advantages and disadvantages.

Scenarios 1 and 2 include reprocessing of the fuel (chemical or mechanical), while scenario 3 does not consider fuel reprocessing. The determination of

TABLE 5-3

COMPARISON OF EXTRAPOLATED COMMERCIAL HYBRID BLANKET CHARACTERISTICS WITH THOSE OF TYPICAL POWER REACTORS

	REACTOR TYPE			
	PWR W (19)	BWR (GE) (19)	CANDU (19)	COMMERCIAL† HYBRID
FUEL ROD O.D., cm	0.94	1.25	1.52	1.45
FUEL ROD PITCH, cm	1.25	1.62	1.65	1.60
AVERAGE CORE/BLANKET POWER DENSITY, W/cm^3	104	56	12.4	66
MAXIMUM CORE/BLANKET POWER DENSITY, W/cm^3	249	120	33	174
AVERAGE HEAT FLUX, W/cm^2	68.5	50.3	50	24*
MAXIMUM HEAT FLUX, W/cm^2	183	112	115	183
MINIMUM DNBR (FLUX RATIO)	1.3	1.9	--	1.2
MAXIMUM FUEL TEMPERATURE, °C	1,788	1,829	1,500	1775
SYSTEM PRESSURE, BARS	155	72	89	56
REACTOR THERMAL POWER, MW	3,411	3,579	1,612	4000
GROSS PLANT THERMAL EFFICIENCY	33.7	33.5	31.0	29 †
GROSS ELECTRICAL POWER OUTPUT, MWe	1,150	1,200	500	1160
AVERAGE BURNUP, MWD/T	33,000	27,500	10,000	9700

+Based on 8.4 $MW\text{-}Yr/m^2$ maximum neutron exposure.
*Assumed fuel shuffling.
†Assumes direct steam cycle.

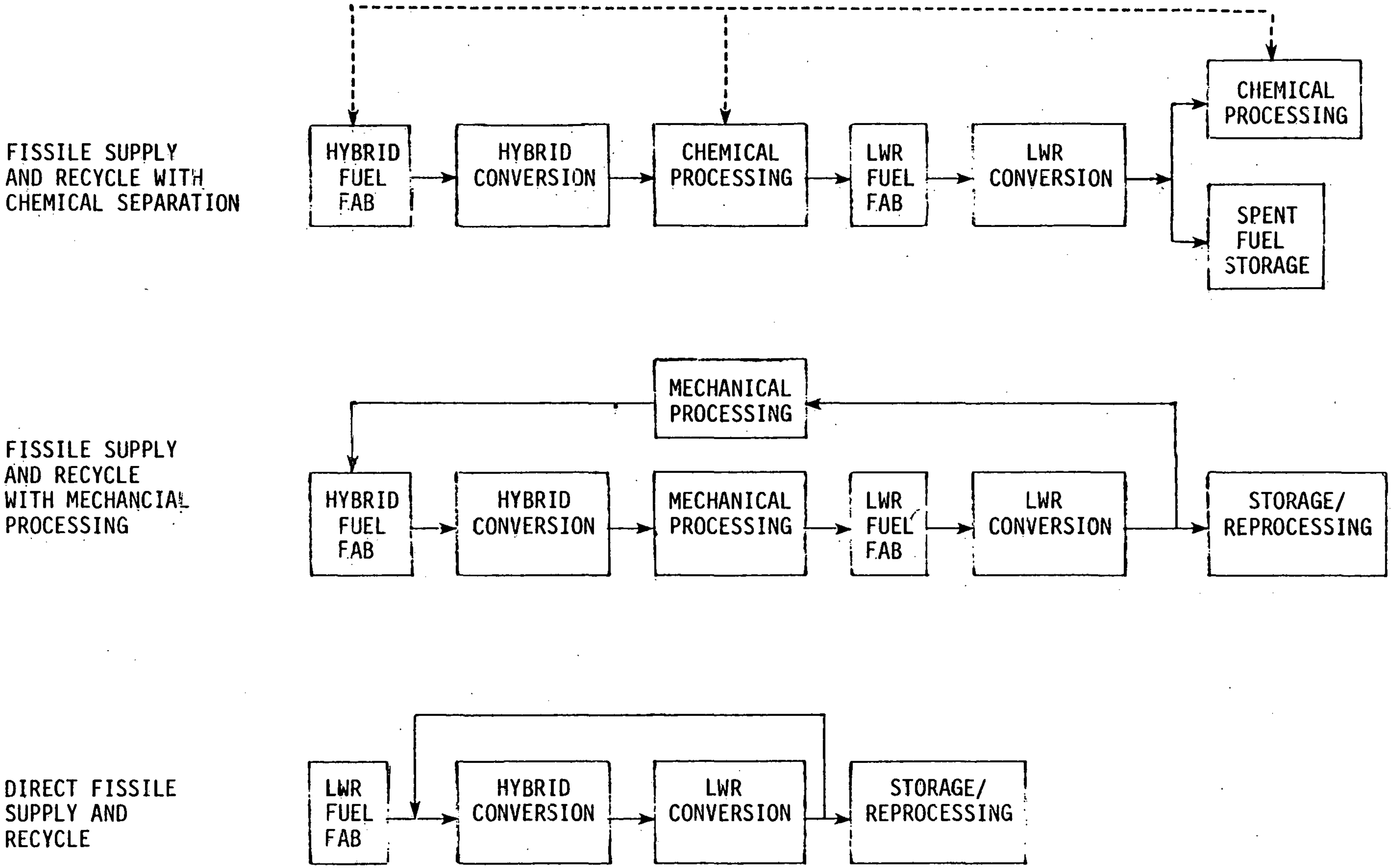


Figure 5-2. Fusion-Fission Hybrid Fuel Utilization Scenarios.

TABLE 5-4

SUMMARY OF HYBRID FUEL RECYCLING ISSUES

OPTION	ADVANTAGES	DISADVANTAGES
CHEMICAL SEPARATION	Permits fuel forms to be optimized for hybrid and LWR individually. Lower hybrid discharge enrichments tolerable, good control over LWR rod feed enrichment.	Requires technology and investment in chemical separation plants. Requires controls to avoid possible diversion of separated fissile material.
MECHANICAL SEPARATION	Permits fuel geometry and cladding to be optimized for hybrid and LWR individually. Re-use of fertile and fissile material without chemical processing. Gaseous fission products can be vented after each cycle. Minimum handling of separated fissile material.	Hybrid and LWR must use same fuel chemically (oxide). Requires hot fabrication and refabrication operations. Possible additional fissile loading required in LWR to compensate for fission product build-up.
DIRECT RECYCLE	Eliminates chemical reprocessing and refabrication steps through one or more cycles. No handling of separated fissile material.	Uncertainty in irradiation limits on fuel and clad. Requires use of same fuel form, clad and geometry in both hybrid and LWR. Requires rods to be vented to avoid fission gas pressure problem. Additional fissile loading required in LWR to compensate for fission product build-up. Uncertainty in range and uniformity of hybrid discharge enrichment.

the optimum (most economic) approach requires extensive comparative economic analyses that include considerations for reprocessing and fuel fabrication/refabrication costs as well as for the feasibility of irradiating spent fuel rods. The basic fuel management scenario selected (with or without reprocessing, chemical or mechanical separation) has significant impacts on the blanket design. For example, if reprocessing is not considered (a refresh cycle), then the fissile enrichment must be compatible with fission reactor requirements and the corresponding neutron exposures and fuel shuffling needed. The required life of fuel clad is another major design consideration. For fuel management scenarios that include reprocessing, then the selection of the fuel form and coolant and the blanket design and operation can be appreciably more flexible.

5.2.2 EFFECT OF COOLANT DENSITY DISTRIBUTIONS

Although the use of two-phase boiling water with radial inflow of the coolant was found to be an attractive option for the DTHR, this approach may not be consistent with optimum commercial operation, where fuel shuffling must be taken into account. This is particularly true if the fuel management scenario inlcudes reprocessing following irradiation in the commercial reactor. This is due to the fact that without reprocessing and initial fuel enrichment, the fissile concentrations (enrichments) required for LWR's can be obtained in a hybrid only by relatively long neutron exposures, With high neutron exposures, the desired fissile concentrations can be obtained only with fast neutrons, i.e. low water number densities such as those attainable with wet or dry steam. However, if a high fissile concentration is to be obtained through reprocessing/separation/and concentration processes, then fuel shuffling at relatively short time intervals can be considered. Under such conditions, a highly thermalized blanket with radial outflow of two-phase, boiling water as coolant may prove to be the optimum mode of operation. This possibility can be deduced from a study of Figures 4-15 and 4-16 where the neutronic performance of blanket with water and two-phase boiling water as coolants are compared. The neutronic performance shown in the figures may be applied to the first 10 cm of the blanket. It can be seen from the figures that at relatively short neutron exposures, (with fuel shuffling thereafter), say up to 3 MW-yr/m^2, the performance of the water cooled blanket is superior to the two-phase cooled blanket in terms of fissile concentration, energy multiplication and tritium breeding. These performances are compared in Table 5-5. The exception is that the net fissile production, averaged over 3 MW-Yr/m^2 is somewhat lower. The greater tritium breeding potential is due to the fact that the neutrons leaking from the back of the blanket can be utilized for tritium breeding. There is clearly a significant advantage with a more thermalized blanket in terms of power production, while a minor penalty is incurred in the fissile production rate. These differences are due to two main factors - the greater neutron moderation of higher density water and the lower parasitic neutron absorption properties of zircaloy relative to stainless steel. These results suggest that a potentially attractive approach is the use of pressure tubes and high pressure water coolant with relatively frequent fuel shuffling/refueling in conjunction with fuel reprocessing. The economics of such an approach should

TABLE 5-5

COMPARISON OF INTEGRATED NEUTRONIC PERFORMANCES AFTER 3 MW-Yr/m^2 OF NEUTRON EXPOSURE

COOLANT	TWO-PHASE, BOILING WATER AT MODERATE PRESSURE	LOW TEMPERATURE, LOW PRESSURE WATER
COOLANT DENSITY DISTRIBUTION	LOW IN THE FRONT OF THE BLANKET, HIGH IN THE REAR	HIGH THROUGHOUT THE BLANKET
BLANKET THICKNESS, CM	40	25
STRUCTURAL MATERIAL	STAINLESS STEEL	ZIRCALOY
FISSILE CONCENTRATION AT THE END OF 3 MW-Yr/m^2, % U-233 IN Th	1.2	1.3
FISSILE FUEL PRODUCTION, AVERAGED OVER 3 MW-Yr/m^2 U-233 ATOMS/FUSION NEUTRON	0.67	0.62
ENERGY MULTIPLICATION AVERAGED OVER 3 MW-Yr/m^2	2.8	7.8*
TRITIUM BREEDING POTENTIAL, AVERAGED NEUTRON LEAKAGE FROM THE BACK OF THE BLANKET, NEUTRONS/FUSION NEUTRON	0.70	0.78

*This blanket thermal power produced is low grade (low temperature, low pressure) and has no capability for power conversion.

be compared with one that needs no fuel reprocessing, but requires longer fuel residence time in the blanket.

Obviously, similar comparisons can be made for other neutron exposures. For example, the fissile production penalty can be eliminated with shorter neutron exposures (shorter than 3 MW-yr/m^2), while it is increased for higher neutron exposures.

These results clearly suggest that for commercial hybrids, where blanket fuel shuffling and fuel reprocessing (separation and concentration) are to be considered, the radial outflow of two-phase boiling water must be evaluated to determine the optimum blanket operating conditions. Since the use of single phase water as the blanket coolant is not attractive overall, radial outflow of two-phase boiling water can be an attractive alternative. This is particularly desirable from the thermal-hydraulic viewpoint for the following reasons:

- Low quality boiling water at the front of the blanket (high rates of heat transfer) can eliminate the need for a separate module front wall coolant circuit.
- High rates of heat transfer (low quality at the blanket inlet) correspond to high fuel rod power densities so that fuel rod temperatures can be reduced significantly or higher rates of power production can be considered.
- Higher tritium breeding potential associated with greater neutron leakage from the back of the blanket as coolant density decreases radially out through the blanket.

It should be pointed out that there are many possible schemes for fuel shuffling for a three-zone blanket, two of which are illustrated in Figure 5-1. Each method and the neutron exposure level provides a unique set of radial power distribution. Since the power density distribution affects the water density distribution, iterative neutronic-thermal hydraulic analyses are required to arrive at a converged solution (See detailed discussion and illustrations in the Appendix). In general, the water density distribution has significant effects on the neutronic and the thermal-hydraulic performance, therefore the iterative analyses are both necessary and important for preliminary blanket designs.

The unique distributions of water number densities for radial inflow of boiling water coupled with the blanket neutronic effects suggest that it may be possible that a self-regulated power-leveling effect exists in the blanket. This may be possible if total flowrate is maintained constant while exit quality is maintained relatively low just after a refueling operation. The higher average water density in the blanket enhances power production. Near the end of the cycle, the exit quality increases with increasing blanket power. The reduced average blanket water density tends to retard burn-up. The net effect could be a reduced amplitude in the blanket power density during each cycle.

5.3 FLOW DISTRIBUTION AND FLOW STABILITY UNCERTAINTIES

Flow distribution in the radial boiling water cooled blanket, two-phase flow instabilities, and corrosion by high quality steam are potentially the most serious technical problems of the reference blanket concept. The coolant headering system for the blanket module was conceived to minimize the use of blanket volume. Forced convective radial coolant flow has an inherent problem with flow distribution because of the effect of gravity and axial pressure drops in the headers (or plena). This could lead to nonuniform axial coolant density distributions. These problems may be solved by engineering design. For example, relatively uniform flow distribution can be attained by flow baffles (provided in the conceptual module design) and by orificing. However, flow stratification problems could still be encountered. Pressure drops in the headers can be minimized by maximizing the size of the headers. The effect of gravity also tends to separate the steam from the water in the exit plenum. The larger the exit plenum, the better is the steam separation. Steam separation in the exit plenum is not necessary; however, because a separate external steam separator would be needed in any case. Uniform flow distrubution is not a serious problem. If swirl vanes or their equivalent are found necessary, the ΔP, coolant pumping power and the structural fraction could increase. Careful attention to engineering and testing to achieve flow and water density uniformity is warranted in a detailed design effort.

Due to the pulsed nature of the blanket power operation, there may be added problems with respect to flow separation/flow instabilities. Again, this represents a detailed engineering problem that can be studied by cold flow and electrically heated flow mock-ups.

5.4 TRITIUM CONTAMINATION

Excessive contamination of the blanket water coolant by tritium, diffusing into the blanket from the plasma region, must be prevented for reasons of safety, economics and corrosion effects. The maximum tritium concentration in steam allowed for direct conversion in a steam turbine is 3×10^{-3} μCi/g.[16] Tritium concentration in water above this value must require either processing to remove the tritium or replacement of the water. Either approach leads to economic penalties because of the high costs associated with the removal of tritium from water, and for the disposal of tritium-contaminated water. This problem can be alleviated by using a vacuum vessel, such as in the DTHR, with its separate coolant circuit. Helium could be the preferred coolant for the vacuum vessel because tritium can be readily removed (relatively inexpensively) from helium.

5.5 DETAILED DESIGN AND BLANKET OPTIMIZATION

The scoping calculations performed to date permit a preliminary definition of a hybrid breeding blanket. For this reason, there was no attempt to optimize the blanket design. In addition, a number of detailed engineering considerations were neglected. For example, hot channel factors, hot spot factors, etc., considerations important in a detailed design analysis, were not taken into account. This is partly due to the lack of established engineering factors for cross flow through a bank of fuel rods and uncertainties in the 2-D and 3-D nonuniformities in nuclear performance. The major reason is that a detailed design and analysis must be based on converged solutions in water and power density distributions and on an optimized blanket design (optimization from the standpoint of overall economics).

There are very limited options for the optimization of a LWR fuel assembly for the hybrid reactor blanket. Some parameters that may be varied are the fuel rod diameter (with the corresponding clad thickness) and the pitch-to-diameter ratio. The effects of these parameters on the blanket design must be evaluated in terms of not only the neutronic performance, but also the blanket pressure and coolant pumping power required. Thus, potential gains in fuel loadings by varying fuel rod diameter and pitch to diameter ratio could be offset by higher structural fractions if coolant pumping power and blanket power production were to be maintained at reasonable levels.

5.6 TRITIUM BREEDING

Although the DTHR blanket concept has no initial provision for tritium breeding, this can be accommodated in a modified blanket design that includes a tritium breeding zone behind the fissile breeding zones. However, the engineering design of such a blanket is not at all straight forward. The contamination of the water coolant could be a problem. The magnitude of the problem depends on the tritium concentration in the water. If it is above 3×10^{-3} μ Ci/g, then a direct steam cycle cannot be utilized. Tritium contamination of the water can be minimized by breeding tritium in separate blanket modules or in modules located only in the inner blanket region, the approach assumed in this study. Lithium water reactions could be avoided by the use of solid lithium compounds. Alternatively, a separate coolant such as helium could be used for the tritium breeding zones, although the use of two different coolants can introduce mechanical design and assembly complexities.

5.7 STRUCTURAL DESIGN CONSIDERATIONS

The DTHR side walls and fuel rod clad were found acceptable at an EOL fast fusion fluence (E>0.1 M_eV, (Q_t) = 1.36×10^{22} n/cm^2) or a neutron flux of 1.2 MW-yr/m^2 in relation to fatigue-crack growth for coolant pressure and through-the-wall temperature differences. Brittle fracture was not considered because of the available ductility in 20% CW-316-SS and zircaloy at the relatively low neutron exposures. However, extrapolations to long term operation with a nuetron exposure of 8.4 MW-Yr/m^2 require the consideration of loss of material ductility induced by irradiation embrittlement. The latter is especially important in protecting against clad fracture and subsequent fission gas release into the coolant caused by power ramps associated with plasma on-off cycling, and leakage of the water coolant through the side walls. An extensive materials data base for 20% CW-316-SS, Zircaloy, and ThO_2 is required at high neutron exposure levels consisting of plane strain fracture toughness (K_{IC}), stress corrosion cracking threshold (K_{ISCC}), fatigue-crack growth (da/dn), and irradiation-creep and swelling relations.

5.8 FUEL ROD VIBRATIONS AND FRETTING

Another major uncertainty in the proposed blanket concept is fuel rod vibration induced by radial two-phase flow of boiling water. This could lead to fuel clad fretting and accelerated corrosion problems. Existing experimental data are not applicable because of the significantly different flow orientations and coolant operating conditions. Consequently, experimental testing would be needed. Such tests can be carried out in conjunction with flow distribution, flow stratification tests. The last can be initiated with flow visualization tests using two-phase freon.

6.0 CONCLUSIONS AND RECOMMENDATIONS

Scoping design calculations have identified boiling water with radial coolant flow as a potentially attractive method for blanket cooling for a DTHR as well as for commercial applications. Neutronic calculations have been performed primarily for radial inflow with single phase water at the blanket inlet and 80% quality steam at the blanket outlet. Although radial outflow is more attractive from the standpoints of thermal hydraulics and maximum power production, the fissile fuel production capability is reduced, primarily because of the presence of the relatively large inlet coolant header with high density water. There is thus a trade-off between fissile breeding and power production. The selection of the optimum mode of coolant flow depends largely on the overall economics of the hybrid reactor, which is dependent on the net electrical power production, the net fissile breeding rate and the amount of tritium that can be bred in the hybrid reactor. For commercial reactors, equilibrium blanket operating conditions must be established. This is achieved through fuel shuffling. Therefore, neutronic analyses with radial inflow of boiling water as coolant and with different fuel management scenarios must be carried out to provide data for comparative economic analyses. A change from boiling water with radial inflow to radial outflow would have little effect on the mechanical designs of the blanket; however, the thermal hydraulic requirements can be relaxed appreciably, because the peak power density region corresponds to maximum heat transfer coefficients and minimum DNB ratio. Therefore, this mode of blanket cooling could provide another design for economic comparison, particularly if a relatively low density coolant (10 to 20% quality) can be obtained for the inlet header to achieve more desirable neutronic performance.

It has been pointed out that for a given neutron exposure, the determination of the blanket neutron performance requires a tandem iterative solution between neutronic and thermal analyses. This is due to the fact that in a two-phase boiling water cooled blanket, the blanket neutronic performance is dependent on the water density distribution. Conversely, the water density distribution is dependent on the power density distribution.

The results presented here have been based on nearly-converged solutions; therefore, they can be considered to be only preliminary. In the case of a blanket design for a commercial hybrid reactor operation, the tandem iterative solutions can be complicated appreciably with fuel shuffling. More precise solutions could be obtained by extensive tandem iterations for each time-step (neutron exposure). Such solutions are justified only for detailed blanket designs once the optimum blanket, fuel management and flow conditions have been established.

For the DTHR, a separate vacuum vessel, separate from the blanket module, is utilized. The vacuum vessel is cooled by water. Because of tritium contamination, the water coolant must operate in a separate closed system. For a similar approach in an EPR design, Maroni's calculations showed that in 2-3 years, the tritium build-up in the water coolant in the vacuum vessel is less than those encountered in existing reactors at Grenoble and Chalk River (CANDU reactors)(17). The tritium contaminated water can therefore be handled in the same manner as in those reactors. The tritium pick-up in the blanket coolant is therefore expected to be far less than that in the vacuum vessel coolant. If the concentration does not exceed 3×10^{-3} $\mu C_i/g$, then a direct steam cycle can be utilized. Otherwise, an external steam generator will have to be included. The same considerations will have to be given when designing a blanket for a commercial hybrid. If a direct steam cycle can be used when a separate vacuum vessel is included (if it is assumed that tritium breeding does not aggravate the tritium contamination problem), then the use of a separate vacuum vessel in a commercial reactor would be highly desirable from the standpoint of power conversion.

It should be emphasized that the reference blanket concept evolved as a result of specific ground rules and design goals. Two of these that have major impacts on the blanket design are 1) the decision to have tritium breeding in separate blanket modules, and 2) the adoption of proven state-of-the-art fission technologies. These two ground rules led to the selection of water reactor technology and its attendant fuel form, clad and coolant. By permitting the breeding of tritium in separate blanket modules, the problems of tritium build-up in the fissile breeding blanket water coolant and incompatibility of water with the potential use of liquid lithium for tritium breeding might be avoided.

The second ground rule imposes a number of design constraints that may encounter difficulties in a commercial application. Among these is the adoption of LWR fuel rod design (long and straight), which lead to difficulties when attempting to provide a relatively high blanket coverage in both the poloidal and toroidal directions and especially in the inner blanket regions, unless the inner blanket region is reserved exclusively for tritium breeding, as was assumed here. The use of shorter fuel rods could extend the wall coverage; however, this would lead to higher fuel rod costs (larger number of fuel rods) unless a simpler approach to fabrication can be developed and justified. A commercial reactor utilizing the approaches and the blanket design adopted here can be considered to be one design for economic evaluation, one that is to be compared with alternate designs based on different ground rules, design goals, and approaches. There are a number of unknown factors and feasibility issues even with the reference concept, a concept based on proven state-of-the-art fission reactor technology and on a simple design approach. This suggests that there is a high degree of risk involved in the development of more complex designs and alternate approaches for near-term demonstration.

6.1 CONCLUSIONS

From the results of this study, the following major conclusions can be made:

- Based on the specific design goals, requirements and constraints delineated, scoping design calculations have identified radial inflow of boiling water as a potentially promising blanket-cooling concept for the DTHR as well as a commercial hybrid.
- The DTHR blanket concept that evolved produces a significant amount of net fissile fuel. With only 24% coverage of the total blanket wall, the DTHR produces a significant amount of net fissile ^{233}U (160 kg(y).
- The DTHR blanket concept has the potential for substantial electrical power production because of the relatively high coolant temperatures and blanket power production. The average blanket power produced over three years with 24% blanket wall coverage is 508 MWt.
- The maximum fuel enrichment attained in the first of the three radially fueled zones in the DTHR blanket is 0.5% in three years (1.2 MW-yr/m^2 exposure).
- The estimated commercial reactor perforamnce based on the DTHR blanket design shows attractive fissile production, enrichments and power production. Therefore, the reference DTHR blanket should demonstrate these capabilities.
- With the use of a separate water cooled vacuum vessel, tritium pick-up in the blanket water coolant does not appear to be a problem; however, detailed anlaysis is recommended in subsequent design studies, particularly if tritium breeding were carried out in separate but adjacent blanket modules.

- Non-uniform axial flow and water densities in radial two-phase coolant flow, two-phase flow instabilities, flow induced rod vibrations and the effects of cyclic operations were identified as potentially the most serious technical problems for the reference blanket concept.
- The reference blanket concept evolved following a specific set of design goals and requirements. Other significantly different blanket concepts can evolve if the blanket ground rules and assumptions were changed. Thus, the reference blanket concept represents a reference with which the economics of alternate concepts can be compared. Other flow orientations and water density distributions are recommended for further study to provide comparisons. In particular, boiling water in vertical flow can provide interesting comparisons. In addition, relatively high pressure dry steam in a conventional pressure tube configuration should be evaluated as well.

6.2 RECOMMENDATIONS

It is recommended that:

- If boiling water in radial flow continues to be an attractive approach, then experimental testing using cold flow and electrically heated blanket mock-ups should be carried out to study and solve the potential two-phase flow problems identified. Such development efforts are required prior to a preliminary conceptual design effort.
- The development of a materials data base is required for Zircaloy, ThO_2, and 20% CW-316-SS prior to a detailed design of a commercial blanket of the type proposed here.
- Some form of inpile tests to examine corrosion effects will also be needed.
- The effect of cyclic operation on fuel rod stress corrosion,

fuel-clad mechanical interaction, ceramic fuel integrity and fuel rod life should also be investigated, although these are phenomena generic to all tokamak-driven hybrid blankets.

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APPENDIX A
PRELIMINARY BLANKET DESIGN CONSIDERATIONS

Following the establishment of the basic design goals, requirements and constraints, the preliminary considerations for power conversion requirements and neutronic scoping calculations, blanket configurations, alternate coolants and coolant flow schemes were considered to determine the relative feasibility and attractiveness of alternate concepts. The goal was to develop a commercial hybrid blanket concpet to be demonstrated in a DTHR. The preliminary feasibility studies and scoping analyses are discusses in this Appendix.

A-1 ALTERNATE COOLANTS

Steam and boiling water for fission reactor cooling have been studied extensively. Recently, Sze et al.[A-1] studied a boiling water cooled Tokamak reactor blanket, while Stevens et al.[A-2] evaluated the use of steam as potential coolant for a non-breeding blanket design. These two alternate approaches to blanket cooling are of interest for the following reasons:

- Compared to single phase water, the lower water densities associated with two-phase water and steam should enhance fissile breeding and reduce fissile burn-up.
- The low temperature, low pressure water cooling system proposed for the DTHR cannot be extrapolated to a commercial reactor if power generation is required.

A study was therefore initiated to assess the viability of steam and boiling water cooled hybrid blankets and compare them with low pressure and high pressure water cooled concepts.

The use of steam as a fission reactor coolant has been studied in a number of countries. The advantages of using steam were well recognized and include the following:

- Well-known chemical, thermodynamic and physical characteristics.
- Extensive experience in designing and manufacturing steam components.
- An expected short development program for all components except the fuel elements because of higher corrosion rates.
- The possibility of using a direct cycle with consequent capital cost economies.

However, there are a number of disadvantages as well:

- High pressure steam necessary for the required high mass flow rates cause increased neutron moderation, which has a detrimental effect on net fissile production.
- The requirement for an extended development and qualification program for a fuel element that can withstand the highly corrosive atmosphere of high-temperature steam.

A-1.1 APPLICATIONS TO TOKAMAK HYBRIDS

These advantages and disadvantages generally apply to a tokamak hybrid blanket as well. However, there are other problems that are unique to a tokamak hybrid blanket application. These are discussed below.

Incompatibility of Fuel Clad

A conceptual design of an Experimental Steam-Cooled Fast Reactor (ESCR) was prepared by the General Electric Company and a group of 14 utilities[A3]. The steam conditions specified were as follows: Inlet steam pressure = 1500 psia; Outlet = 1415 psia; Inlet steam temperature = 610°F, Outlet = 950°F. The most promising fuel clad material identified for these operating conditions was Incoloy-800. This material contains 32% nickel. Nickel-58 has a high absorption cross section that transmutes to nickel-59. Nickel-59 in turn is a prolific producer of helium. These reactions severely limit the useful life of the clad.

High Coolant Pumping Power

The specific heat of steam increases with increasing temperature and pressure. At relatively low temperatures and pressures, its specific heat is approximately one half of that of water. Therefore, given the same amount of blanket thermal power, the mass flow rate, pressure drop, blanket pressure and coolant pump work would all be appreciably greater when steam is used as coolant. Thus, relatively high blanket pressures would be desirable to reduce steam pumping power. This results in relatively high steam densities which could obviate the neutronic advantages associated with low density steam.

Several possible modes of steam-water operation can be considered here:

- Saturated steam in, superheated steam out;
- Subcooled water in, saturated steam out;
- Subcooled water in, two-phase mixture out to steam separator.
- Two-phase water in, two-phase water out;

Neutronic performance should favor the first mode if the ratio of steam to fuel volume fractions can be kept the same as the other two modes; otherwise there may be no net advantage. Options 2 and 3 should provide some neutronic performance improvements over the all-water coolant case, because of the lower water number densities. However, the distribution of water number densities is nonuniform in the vertical direction unless radial inflow and outflow are considered. In addition, all three options may require high temperature and high pressure operations that will increase the ratios of the coolant and structures volume fractions to the fuel volume fractions.

Because the space in the blanket region of a tokamak is at a premium, forced circulation will be adopted for boiling water coolant to avoid the need for a chimney in a natural circulation system. The nonuniform radial power distribution can cause difficult design problems because this tends to cause nonuniform flow distributions unless several separate flow circuits are created.

Axial two-phase flow is accompanied by nonuniform axial water number densities, nonuniform axial power densities, and nonuniform axial fuel enrichments. This can result in the requirement for complex iterative neutronic-thermal mechanical design calculations and two-dimensional neutronic calculations.

A-2 ALTERNATE BLANKET CONCEPTS CONSIDERED

A sequence of preliminary blanket concepts was evaluated as depicted schematically in Figure A-1 together with their major operating characteristics, advantages and limitations. The DTHR water cooled, baseline blanket concept is included to provide a comparison basepoint. The sequence of concepts represent increasing deviations from state-of-the-art LWR technology.

A-2.1 BLANKET CONCEPTS WITH VERTICAL COOLANT FLOW

The first alternate blanket configuration considered in the development of an improved concept consisted of fuel assemblies in 8 cm x 8 cm square modules $\sim$ 4 meters long with vertical coolant flow (concepts 2 and 3). This module can accommodate typical LWR and BWR fuel assemblies. Dry steam was initially considered as the coolant because of the desire to increase the fuel enrichment (concept 2). Neutronic calculations for the water-cooled blanket showed that the presence of high density water caused significant neutron moderation and softening of the spectrum so that significant fissile burn-up occurred, making it difficult to achieve reasonable fuel enrichment goals. Calculations using helium as coolant showed the opposite effect. These results suggested that the use of dry steam as coolant would appreciably increase the fissile enrichment.

However, there are a number of neutronic, thermal-hydraulic and mechanical design difficulties that rendered the concept highly unattractive. The major limitation is that the coolant pumping power would be prohibitive. Coolant pumping power can be reduced by increasing the steam pressure and the steam temperature rise. Increased steam pressure would require increased module wall thickness and reduced spacing between walls, resulting in increased structural volume fractions. As an illustration, the set of blanket operating conditions given in Table A-1 yields a steam pumping power to blanket power ratio of 1.68, which is clearly unacceptable.

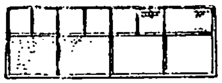
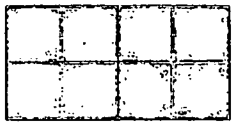
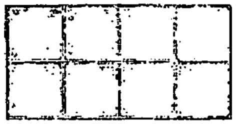
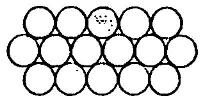
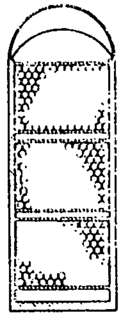
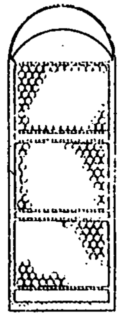
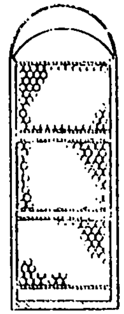
	1	2	3	4	5	6	7
BLANKET CONCEPT GENERAL DESCRIPTION	RECTANGULAR MODULES BASELINE CONCEPT	SQUARE BLANKET MODULES	SQUARE BLANKET MODULES	CONVENTIONAL PRESSURE TUBE MODULE	SPLIT CYLINDRICAL HEAD BLANKET MODULE	SPLIT CYLINDRICAL HEAD BLANKET MODULE	SPLIT CYLINDRICAL HEAD BLANKET MODULE
BLANKET CROSS-SECTION							
COOLANT	WATER	DRY STEAM	TWO PHASE WATER	SINGLE PHASE WATER	TWO PHASE WATER	TWO PHASE WATER	TWO PHASE WATER
FLOW ORIENTATION	VERTICAL	VERTICAL	VERTICAL	VERTICAL	RADIAL OUTFLOW	RADIAL INFLOW	RADIAL INFLOW
INLET STEAM QUALITY	0.0	1.0	0.0 (SATURATED WATER)	0.0	0.0 (SUBCOOLED WATER)	0.0 (SUBCOOLED WATER)	0.0 (TWO PHASE WATER)
EXIT STEAM QUALITY	0.0	1.0	1.0 (SATURATED STEAM)	0.0	1.0 (SATURATED STEAM)	1.0 (SATURATED STEAM)	0.8 (SUBCOOLED WATER)
MAJOR ADVANTAGES	• LOW TEMPERATURE, LOW PRESSURE, LOW COST • HIGH FUEL LOADINGS • HIGH FUEL/STRUCTURE RATIO • RELATIVELY NEUTRONIC PERFORMANCE • RELIABLE STRUCTURAL DESIGN	• RELIABLE STRUCTURAL DESIGN	• RELIABLE STRUCTURAL DESIGN • RELATIVELY LOW HOT ROD TEMPERATURE	• RELATIVELY HIGH POWER CONVERSION CAPABILITY • REASONABLE COOLANT PUMPING POWER	• RELATIVELY HIGH POWER PRODUCTION • RELATIVELY LOW COOLANT PUMPING POWER • RELATIVELY LOW HOT ROD TEMPERATURES	• RELATIVELY HIGH FISSILE CONCENTRATION • RELATIVELY HIGH FISSILE PRODUCTION • RELATIVELY LOW COOLANT PUMPING POWER	• RELATIVELY HIGH FISSILE CONCENTRATION, • RELATIVELY HIGH FISSILE PRODUCTION • RELATIVELY LOW COOLANT PUMPING POWER
MAJOR DISADVANTAGES	• NO ELECTRICAL POWER PRODUCTION CAPABILITY	• EXCESSIVE COOLANT PUMPING POWER FOR REASONABLE POWER DENSITIES • MAXIMUM NEUTRON EXPOSURE AND POWER DENSITY LIMITED BY COOLANT PUMPING POWER	• RELATIVELY LOW FISSILE PRODUCTION • NON-UNIFORM AXIAL AND RADIAL POWER AND WATER DENSITY DISTRIBUTIONS UNLESS COMPLICATED FLOW SYSTEMS WERE DEVISED • RELATIVELY HIGH PRESSURE DROP, MAXIMUM NEUTRON EXPOSURE LIMITED BY Δ P	• RELATIVELY LOW FUEL VOLUME FRACTIONS • RELATIVELY LOW FUEL TO STRUCTURE RATIOS • RELATIVELY LOW FISSILE PRODUCTION	• RELATIVELY LOW NET FISSILE PRODUCTION DUE TO MODERATION OF 14 MeV NEUTRONS • RELATIVELY LOW FISSILE CONCENTRATIONS	• MAXIMUM NEUTRON EXPOSURE AND POWER DENSITY LIMITED BY HOT ROD TEMPERATURE CONSTRAINTS	• MORE DIFFICULT STRUCTURAL AND MECHANICAL DESIGN

Figure A-1. Alternate Blanket and Coolant Flow Schemes Evaluated

Another major disadvantage associated with vertical coolant flow is the large peak to average blanket power ratio. This leads to relatively low average blanket exit coolant temperatures and reduces the power conversion efficiency.

The coolant pumping power can be reduced by using boiling water as coolant with either wet or dry steam (concept 3) exiting from the blanket. Calculations for 33.3 bars coolant pressure and an exit steam quality of 30% yielded reasonably low coolant pumping power (the ratio of coolant pumping power to blanket thermal power for a neutron exposure of 0.8 MW-yr/m^2 was only 0.003). However, the total pressure drop calculated was 130 psi. The high coolant pressure drop clearly presents a problem, since it leads to high module internal pressures and increased module wall thicknesses. Moreover, the problem is aggravated with increasing neutron exposure because the blanket power increases with increasing irradiation. As an illustration, for a neutron exposure of 4 MW-yr/m^2, the pressure drop and the ratio of coolant pumping power to blanket power would be increased by roughly a factor of 25 over the 0.8 MW-yr/m^2 case. While the coolant pumping power remains tolerable (ratio of pumping power to blanket thermal power is 0.075), the blanket internal pressure and the module wall thickness required would be prohibitive. Another major disadvantage for this flow configuration is the nonuniform axial coolant density distribution. This leads to nonuniform axial neutronic performance, complicating the blanket design and analysis and fuel management. Thus, the vertical flow, boiling water-cooled blanket concept does not appear to be attractive for a commercial hybrid reactor. An obvious solution to the coolant pressure drop and pumping power problem was the consideration of radial flow, because the flow cross-sectional area in the radial direction is significantly greater, while the coolant flow path is shortened appreciably.

A-2.2 PRESSURE TUBE CONCEPTS

It was apparent that the use of pressure tubes in the blanket could eliminate the coolant pressure and pumping power problems because relatively thin-walled pressure tubes can be used with relatively high coolant pressures. The latter increases the coolant density and reduces the pumping power.

TABLE A-1
MODULE OPERATING CONDITIONS

NEUTRON EXPOSURE, $MW\text{-}Yr/m^2$	0.8
MAXIMUM FUEL CLAD (ZIRCALOY-4) TEMPERATURE, °C	538
INLET STEAM PRESSURE, BARS	33.3
INLET STEAM TEMPERATURE, °C	242
MEAN STEAM TEMPERATURE RISE, °C	180
MEAN BLANKET EXIT STEAM TEMPERATURE, °C	422
MAXIMUM, HOT CHANNEL EXIT STEAM TEMPERATURE, °C	482
HOT CHANNEL FILM TEMPERATURE DROP, °C	56
RATIO OF STEAM PUMPING POWER TO BLANKET POWER	1.68
MODULE PRESSURE DROP, BARS	8.7

Figure A-2 is a schematic diagram of three conventional pressure tube configurations. The different configurations were investigated to determine their effects on the overall blanket composition when arranged in several rows as shown in Concept 4 of Figure A-1.

Parametric analysis was carried out to evaluate the effect of coolant pressure in the pressure tube wall thickness and blanket composition with the overall blanket thickness held roughly constant. Two sets of conditions were evaluated. One corresponds to a constant peak coolant steam pressure of 2000 psia while the second corresponds to a constant pressure tube wall thickness of 1.7 cm. The results of the analysis are summarized in Table A-2. It is clear from these results that the major penalty with the use of conventional pressure tubes is the low fuel volume fractions and low fuel-to-clad/structure volume fraction ratios, where the pressure tubes were assumed to be made of stainless steel. This penalty is more evident when the blanket composition in Table A-2 is compared with those of a typical square or rectangular pressure vessels, as shown in Table A-3.

It is clear from this comparison that a rectangular pressure vessel produces the higher fuel volume fraction as well as a higher fuel to structure ratio. Thus, this blanket configuration should be preserved. However, the use of steam as a coolant results in excessive pumping power. An examination of the pumping power equation shows that the coolant pumping power is directly proportional to the length of the flow path and inversely proportional to the flow cross-sectional area. This suggests that if maximum fuel volume fractions were to be preserved while minimizing coolant pumping power, the coolant should be introduced into and removed from the blanket radially. This accomplishes two things: the coolant flow path is reduced significantly (from about 5 meters to approximately 0.8 meter) while the coolant flow area is increased appreciably (by a factor of approximately 3). Thus, for the same coolant temperature rise, coolant pumping power can be reduced by a factor of $\sim$ 200.

It should be emphasized that the pressure tube concept can be made technically feasible. However, the rectangular pressure vessel leads to more efficient use of blanket volume and hence lower costs per unit mass of fissile or unit power produced.

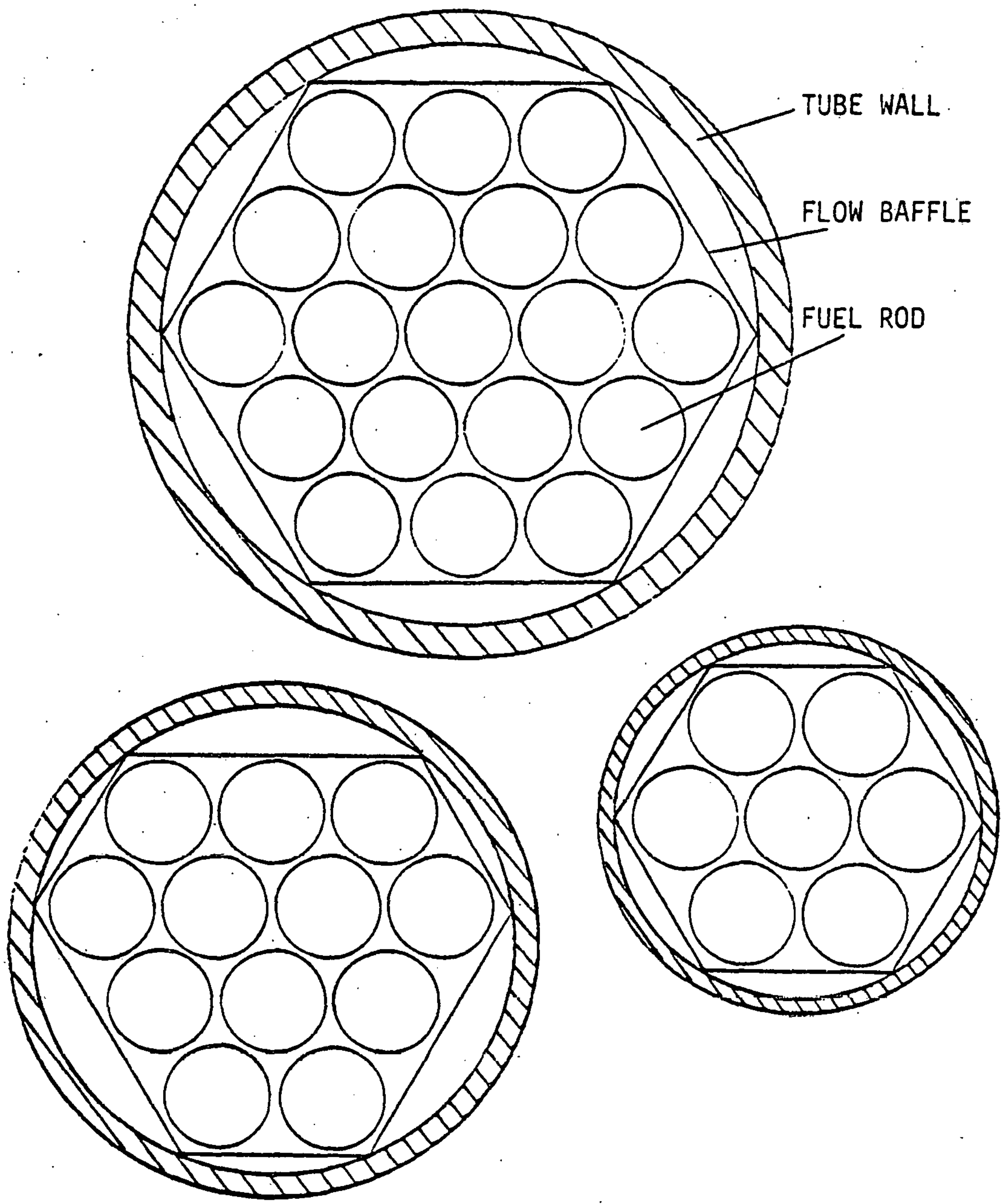


Figure A-2. Conventional Pressure Tube Configurations Evaluated.

TABLE A-2
COMPARISON OF BLANKET COMPOSITIONS

Type of Comparison	Fixed, Constant Pressure			Fixed, Constant P-Tube Wall Thickness		
Number of Rods/tubes	7	12	19	7	12	19
Tube inside diameter, cm	5.2	6.8	8.4	5.2	6.8	8.4
Tube wall thickness, cm	0.70	0.96	1.56	1.70	1.70	1.70
Tube outside diameter, cm	6.60	8.72	9.96	8.60	10.20	11.80
Number of pressure tubes	59	30	26	30	26	17
Number of fuel rods	413	360	494	210	312	323
Maximum steam pressure, psia	2000	2000	2000	2900	2650	2000
Volume Fractions						
Fuel	.238	.215	.266	.122	.176	.182
Coolant	.240	.212	.248	.124	.175	.170
Zircaloy	.372	.350	.290	.500	.528	.419
Void	.150	.223	.196	.254	.121	.229
	1.000	1.000	1.000	1.000	1.000	1.000
Steam Density for $\Delta T_{cool} = 200^{o}F$, lb/ft^3	3.84	3.84	3.84	8.72	5.44	3.84
Maximum Clad Temperature, ^{o}F	965	986	940	1070	1039	996
Steam pumping power to blanket power ratio	0.145	0.077	0.007	0.001	0.0024	0.007

TABLE A-3

COMPARISON OF BLANKET COMPOSITIONS

BLANKET MODULE CONCEPT	RECTANGULAR PRESSURE VESSEL		CONVENTIONAL PRESSURE TUBES	
GENERAL CONFIGURATION	FUEL RODS IN RECTANGULAR BLANKET MODULE PRESSURE VESSEL		19 FUEL RODS/PRESSURE TUBE SEVERAL PRESSURE TUBES PER MODULE	
	OVERALL COMPOSITION	COMPOSITION OF FUEL LATTICE ONLY	OVERALL COMPOSITION	COMPOSITION OF FUEL LATTICE ONLY
FUEL VOLUME FRACTION	0.44	0.52	0.27	0.47
COOLANT VOLUME FRACTION	0.31	0.36	0.25	0.43
ZIRCALOY CLAD VOLUME FRACTION	0.10	0.11	0.06	0.10
SS STRUCTURE VOLUME FRACTION	0.14		0.23	
VOID VOLUME FRACTION	0.01	0.01	0.19	
FUEL/SS RATIO	3.14		1.17	

Advanced Pressure Tubes

An advanced pressure tube concept, with potential for improved fuel loadings, was also considered. This consisted of coextruded metallic thorium-zircaloy clad fuel rods (with central coolant channels) of the type developed at BNWL and SRL[(A-4)]. Analysis showed that the fuel volume fraction can be theoretically increased to 38%. However, the critical flaws with this concept is the relatively low heat transfer surface area to volume ratio and the relatively high fuel swelling with burnup for metallic fuels. In order to contain the resultant pressure, the inner and outer concentric clads must be thickened appreciately even if zircaloy were replaced with stainless steel. The result is that a realistic fuel rod design of this type would lead to significantly reduced fuel loadings and to high stainless steel fractions. The low heat transfer surface areas and the need to limit the outer clad temperature to allowable design values result in relatively low coolant temperatures and high flow rates/pressure drops/coolant pumping power. Comparison with parallel coolant flow outside fuel rods of the same outside diameter as the advanced pressure tube, the same cladding thickness and the same allowed peak clad temperature, showed that the advanced pressure tube concept requires coolant pumping power that is $\sim$ 3 times greater than that for the PWR fuel assembly. This was true even though the coolant pressure for the pressure tube was increased three fold. Meanwhile, the maximum allowed coolant temperature (exit) is reduced by 50°C for the advanced pressure tube concept.

Alternatively, appreciably smaller fuel rods can be considered (smaller than those developed by BNWL and SRL) in order to maintain higher coolant temperature and meet the clad temperature constraint. This is due to the fact that the heat transfer surface area is reduced appreciably for the internal coolant channels, while the peak clad temperature is found on the external clad. Additional external cooling could be considered; however, this would simply eliminate the primary advantage of the concept.

The above considerations, coupled with the relatively poorly developed technology for coextruded fuel rods compared to standard LWR fuel rods, suggested that the concept not be considered for the DTHR blanket.

A-2.3 BLANKET CONCEPTS WITH RADIAL COOLANT FLOW

Radial coolant flow (cross flow, perpendicular to the fuel rods) was investigated next. As depicted in Table A-1, dry steam was the first coolant considered. This was found to present hot rod and module front wall cooling problems. The latter is due to the relatively poor heat transfer characteristics of dry steam (relative to boiling water) and the high power densities associated with the hot rods. The neutronic and thermal hydraulic considerations led to the evaluation of boiling water and wet steam as blanket coolant in various radial flow schemes. Radial outflow of boiling water was found to be ideal from the thermal hydraulic standpoint, because both the hot rods and the module front wall can be adequately cooled by subcooled water in either forced convective surface or bulk boiling modes of heat transfer. However, the best neutronic performance from the standpoint of maximizing fissile breeding and fissile enrichment was attained with one-pass radial inflow of boiling water. This flow scheme provided low density mixtures of steam and water at the front of the module facing the plasma neutrons and increasing water densities away from the plasma. This water density distribution produces the most favorable neutronic effects when fuel shuffling is not considered: Low water number densities in the front of the blanket minimized the moderation of 14 MeV neutrons to permit high neutron multiplication and efficient fissile breeding. The increasing water densities towards the outer module zones moderate the neutrons to minimize neutron leakage and enhance power production. Because the maximization of power production is not an important goal for the DTHR, radial coolant inflow was selected as the mode of coolant flow.

A-3 PARAMETRIC ANALYSIS FOR REFERENCE BLANKET CONCEPT

Parametric and iterative thermal and neutronic analyses were performed in an effort to determine the following:

- The maximum coolant pressure allowed based on thermal, mechanical and structural design constraints.
- The minimum coolant density (at the front of the module) consistent with thermal, hydraulic and mechanical constraints.
- Trade-offs between fuel loadings/fissile breeding and electrical power production.
- The effect of neutron exposure on coolant and blanket operating conditions.
- Determine the maximum exit steam quality permitted to provide adequate hot rod cooling such that the peak hot rod clad (zircaloy) temperature is maintained below 300°C.

A-3.1 COOLANT AND POWER DENSITY DISTRIBUTIONS

Preliminary thermal hydraulic scoping calculations utilized the power density distributions calculated based on dry steam as coolant throughout the blanket. This power density distribution was used to calculate a second approximation to the coolant density distribution using boiling water as coolant. This was then used in a second set of neutronic calculations to produce a new set of power density distributions. A second iteration on thermal hydraulic calculation was then made, based on the new set of power density distributions to produce a third set of coolant density distribution, etc. The successive power density and coolant density approximations are shown in Figures A-3 and A-4. The differences in blanket thicknesses modeled reflect considerations of the effects of neutron leakage: a relatively leaky blanket required a relatively thick blanket in order to maintain efficient neutron utilization. These figures

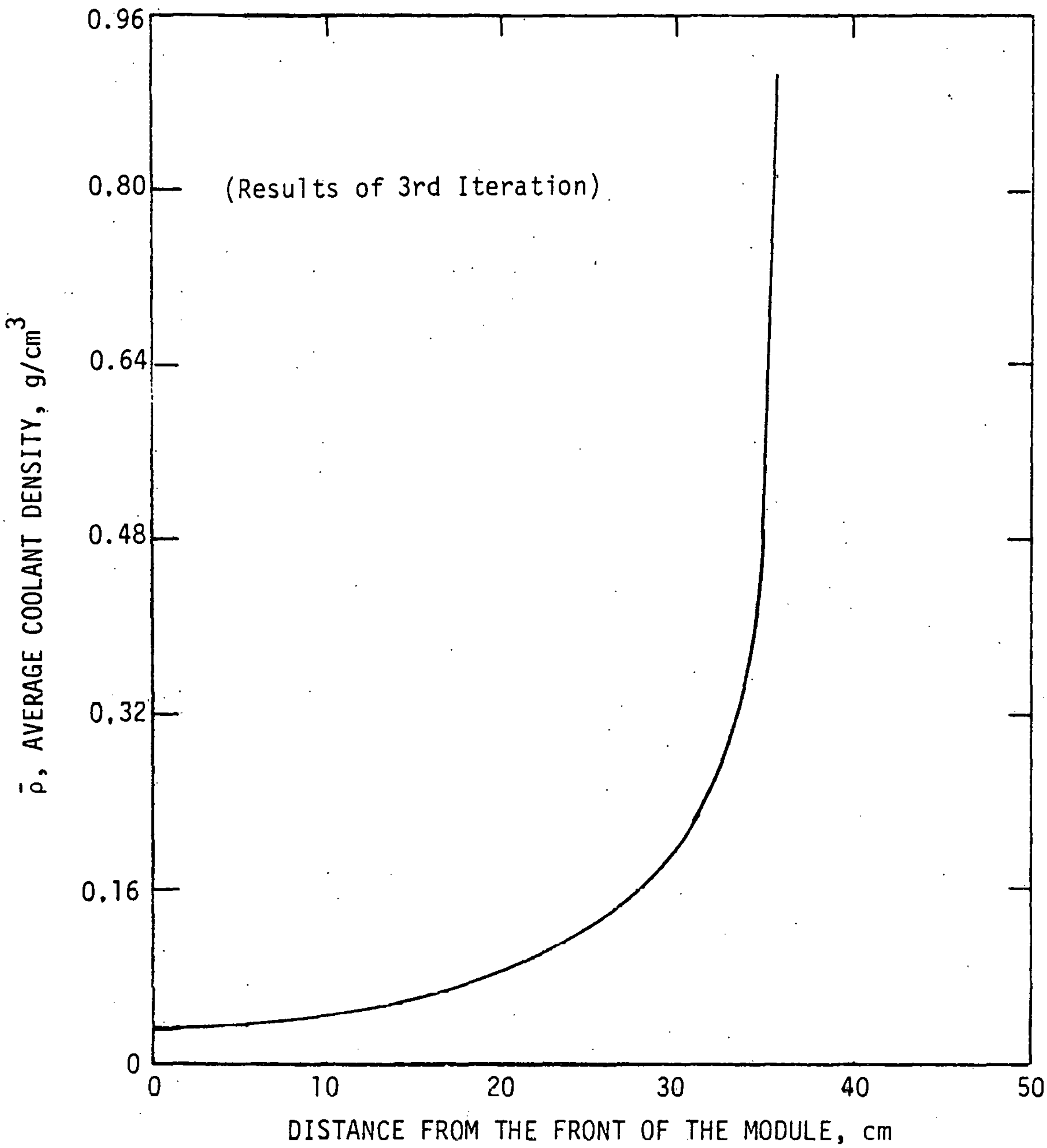


Figure A-3. Radial Water Density Distribution for Radial Inflow of Boiling Water. 33.3 Bars Outlet Pressure, $X_e = 0.80$

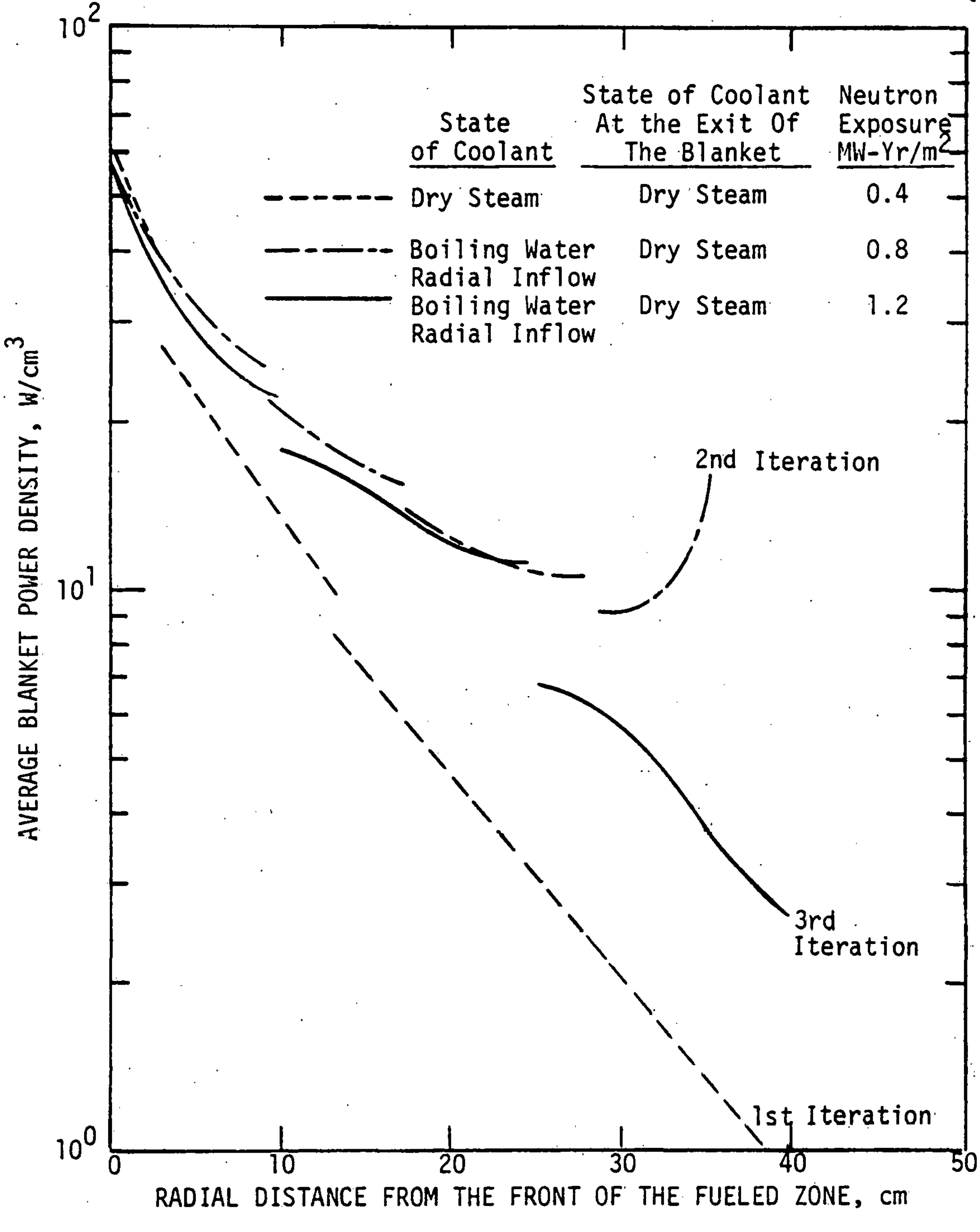


Figure A-4. Power Density Distributions.

also show that convergences in both coolant density and power density distributions are approached rapidly. It should be pointed out that the power density distribution shown in Figure A-4 were for different neutron exposures. This was done to minimize computations and still permit the evaluation of the effects of neutron exposure with nearly converged design solutions. The use of different neutron exposures in the iterations turned out to be of little consequence from the standpoints of thermal, hydraulic and mechanical designs, because the hot rod and the average blanket power densities did not change appreciably and because the hot rod generally controlled the blanket design. The power density distributions shown in Figure A-4 were based on approximately equal hot rod (peak) power densities.

A-3.2 BLANKET PRESSURES AND COOLANT TEMPERATURES

Because of the radial flow orientation, forced convective boiling and vaporization (vs natural circulation) is the most practical approach. If the coolant pressure drop through the blanket is neglected as a first approximation (subject to verification) the peak coolant temperature (with the assumption of no superheat) at the exit of the blanket is simply the saturation temperature for any given pressure. For both radial inflow and outflow, this assumption is conservative. In the first case, coolant pressure drop causes a reduction of the saturation temperature at the exit end of the blanket where the hot rods are located. Significant subcooling of the inlet water can be considered. Thus, the maximum coolant temperature was based on the saturation temperautre corresponding to the maximum coolant inlet pressure. This is shown in Figure A-5 for the range of pressure of interest to DTHR blanket design.

A-3.3 STRUCTURAL REQUIREMENTS AND BLANKET COMPOSITIONS

The blanket module with the bullet shaped cross section was evaluated to determine the effect of the coolant pressure on the module structural requirements and the blanket composition. The primary purpose was to identify reference coolant operating conditions and preliminary mechanical - structural requrements.

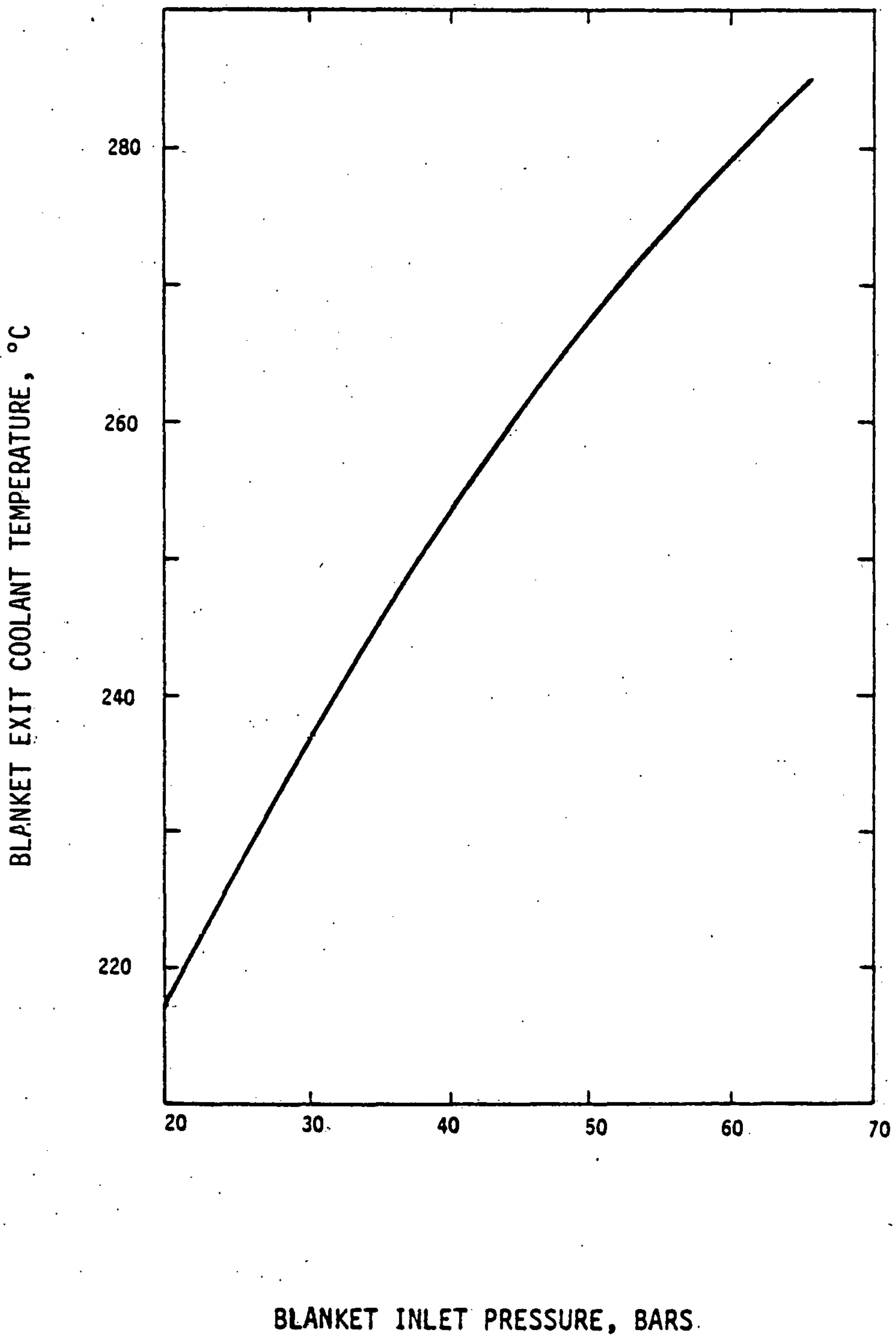


Figure A-5. Effect of Blanket Inlet Pressure On Outlet Coolant Temperature

The boundary conditions for this preliminary analyses were as follows:

- The structural material is solution annealed 316-SS
- The design criteria is by Section III of the ASME Boiler and Pressure Vessel Code for Nuclear Reactors. This criteria is not applicable to detailed analysis; however it provides a convenient criteria for preliminary scoping analysis.
- The radial depth of the module was assumed to accommodate approximately 40 cm of fuel rods. The width of the module was specified to accommodate a maximum of 12 rods on a p/d ratio of 1.1. The reference fuel rod has an OD of 1.45 cm.

Due to the maximum temperature limitations for the fuel rod cladding and module structural material, the maximum two phase coolant pressure which would be applicable, because of the pressure associated saturation temperature would be 53 bars. The thermal and mechanical loads on the blanket are due to the nuclear heat deposition and the internal pressure of the coolant. The mechanical load was assumed to be constant over the life time of the module, while the thermal load is transient, following the plasma on-off cycle. The magnitude of the nuclear heat deposited is a function of radial depth in the blanket. The upper limit was estimated by nuclear analysis to be 13 W/cm^3.

The structural analysis of the module was directed towards the calculation of the stresses developed during the worst loading period, i.e. during the plasma burn. The stresses in the front wall, split cylindrical section, are due to the coolant pressure and the thermal load. The pressure stress included a membrane hoop stress. The stresses in the side walls are due to a bending stress (from the coolant pressure acting normal to the unsupported side wall), a membrane stress (developed due to the coolant pressure acting at either end of the module), and the thermal stress (induced by the temperature gradient acting through the wall). The overall structure is limited by the secondary thermal stresses which are functions of the wall thickness.

The supporting perforated stiffening walls are sized to withstand uniaxial force induced from the coolant pressure acting on the side walls. The limiting constraint in determining the pitch diameter of the coolant holes is the maximum allowable temperature. The problem was modeled by assuming that the region surrounding each circular coolant channel constitutes an insulated cylinder with a uniform heat source having an internal convective boundary. The worst possible loading occurs on the stiffening plate nearest the front wall. The volumetric heating was taken to be 13 w/cm^3 and the coolant has the highest quality at this point. The stiffening plate necessitated a pitch diameter ratio of 1.85 which yields a plate that has a 42% void fraction.

Figures A-6 and A-7 illustrate the trends in blanket material composition versus coolant pressure. At low coolant pressures, the fuel volume fraction increases while the structural volume fraction decreases. However, the low pressures yield a highly inefficient thermodynamic power cycle. To increase the performance of the electrical power producing capabilities of the blanket, the design must be able to withstand high coolant pressures. This can be attained at the expense of a reduced fuel volume fraction.

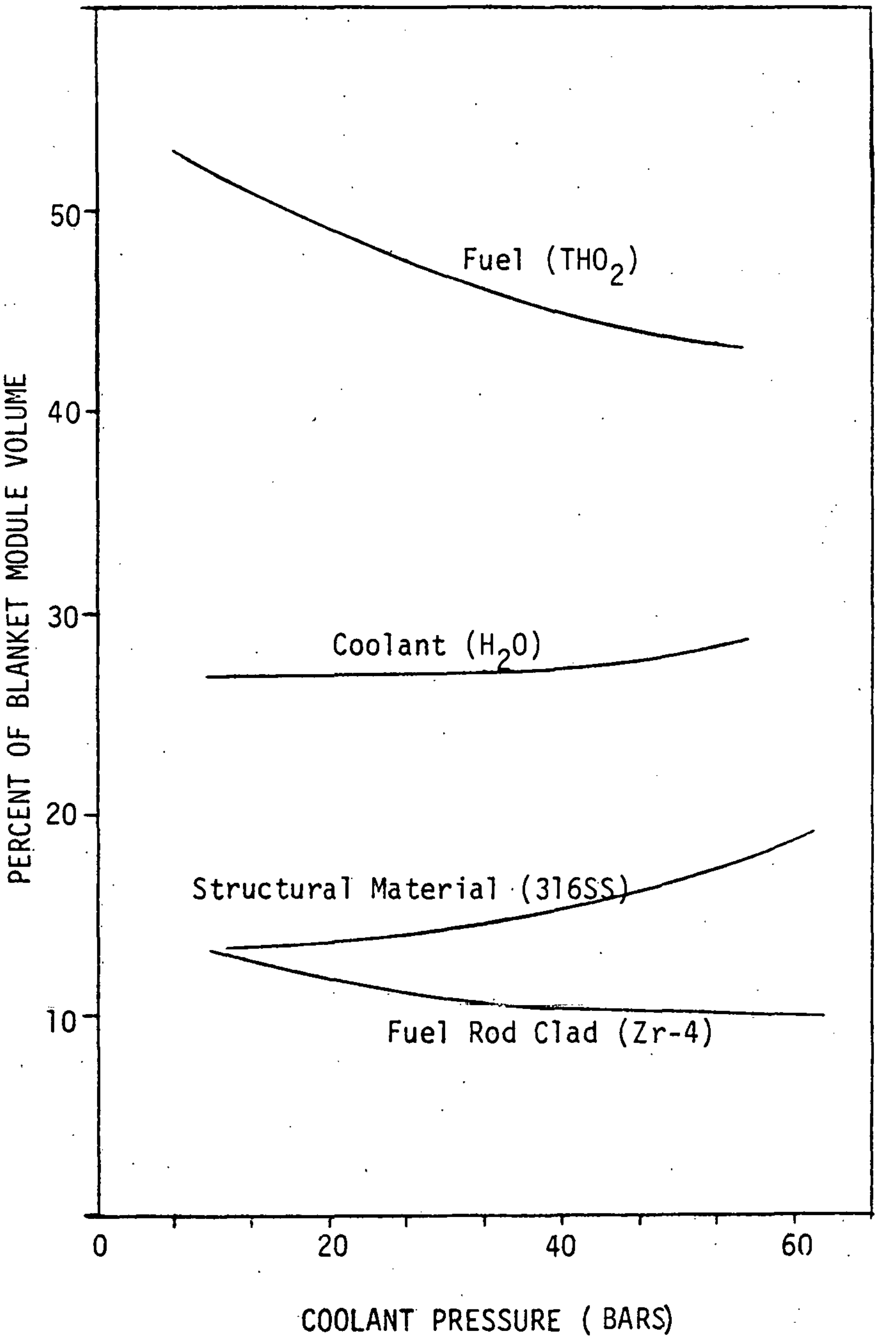


Figure A-6. Effect of Coolant Pressure on Blanket Composition.

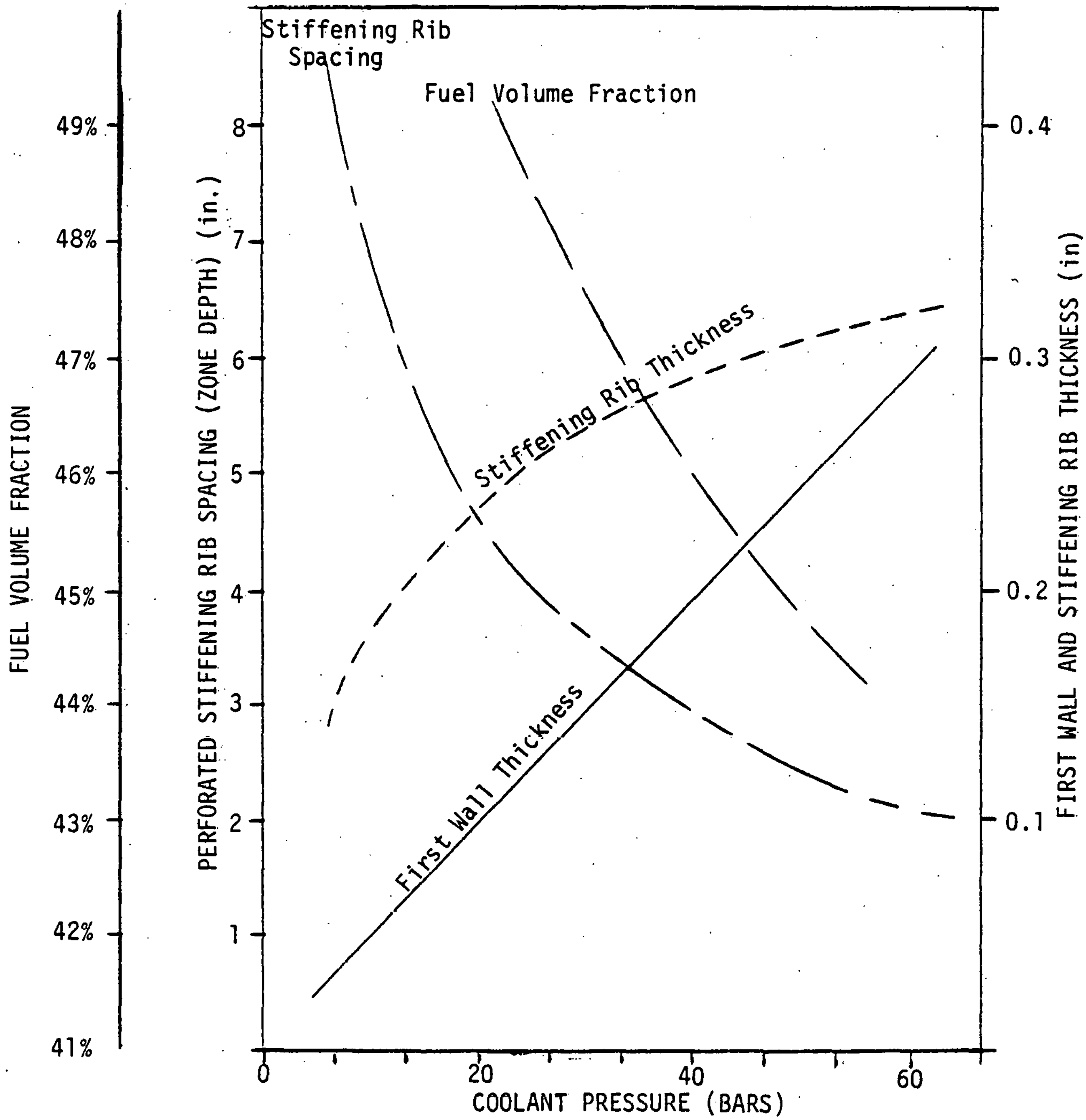


Figure A-7. Effect of Coolant Pressure on Blanket Design.

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