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EVALUATION OF STEAM AS A POTENTIAL COOLANT FOR NONBREEDING BLANKET DESIGNS

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# EVALUATION OF STEAM AS A POTENTIAL COOLANT FOR NONBREEDING BLANKET DESIGNS*

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A steam-cooled nonbreeding blanket design has been developed as an evolution of the Argonne Experimental Power Reactor (EPR) studies. This blanket concept complete with maintenance considerations is to function at temperatures up to 650°C utilizing nickel-based alloys such as Inconel 625. Thermo-mechnical analyses were carried out in conjunction with thermal hydraulic analysis to determine coolant channel arrangements that permit delivery of superheated steam at 500°C directly to a modern fossil plant-type turbine. A dual-cycle system combining a pressurized water circuit coupled with a superheated steam circuit can produce turbine plant conversion efficiencies approaching 41.5%.

## INTRODUCTION

A number of alternate blanket designs and coding concepts are under investigation at Argonne National Laboratory for the tokamak-type fusion power reactors.[1,2] Most of these studies addressed the relative merits of a wide range of coolants (e.g., liquid lithium, helium, and pressurized water) and structural materials (e.g., austenitic stainless steel and vanadium-based alloys). As the first generation of fusion power plants are likely to be nonbreeding types, we have extended our studies to evaluation of steam[3] as a potential coolant with stainless steel and more recently with nickel-based alloys such as Inconel 625 as the blanket material.

The principal functions of the proposed blanket systems for these reactors are to provide a relatively thin zone (30-50 cm) of high-density shielding material that can function at elevated temperatures (up to 650°C) in a way that permits the retrieval of most of the thermalized neutron and gamma energy as sensible heat. Conceptual design and analysis of potentially useful blanket configurations that could perform these functions has been a major undertaking within the EPR studies conducted to date. A significant drawback of many of the previous conceptual design and scoping studies of tokamak-type fusion power reactors[1,2,5,6] is that they have not taken into consideration the effect of thermal-hydraulic design on the thermal stress distribution within the blanket block. From a purely thermal-hydraulic point of view, the conceptual design of the blanket is oriented toward obtaining maximum coolant temperature for the thermodynamic cycle without exceeding the allowable temperature in the structural material of the blanket block. It can be shown from mechanical response analysis that severe thermal stresses can be present within the blanket blocks when the coolant channel distribution is selected to meet only the thermal hydraulic criteria and the stress concentrations are highly sensitive to coolant channel arrangement. Hence, thermo-mechanical analyses were carried out in conjunction with thermal-hydraulic analyses in selecting the coolant channel arrangements.

## DESCRIPTION OF EPR BLANKET/SUPERHEATER

The specific design proposed is an extension of the earlier ANL/EPR design[1,3] wherein the inner and outer blanket segments are made of stainless steel and cooled by pressurized water while the horizontal upper and lower blanket blocks (superheater regions) are made of Inconel 625 and are cooled by steam (see Fig. 1). The use of

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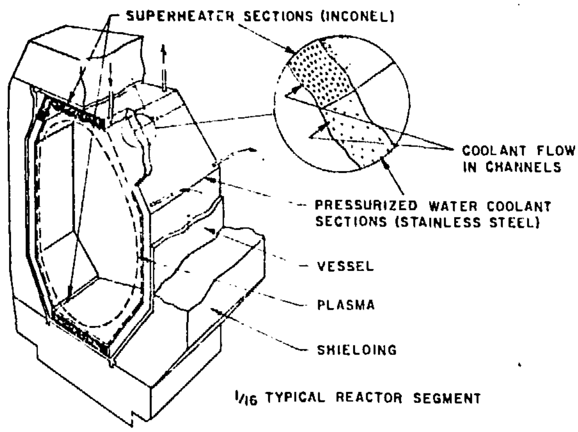


FIGURE 1. Schematic of EPR blanket superheater.

Inconel 625 with improved high-temperature material properties, 650°C versus 550°C for stainless steel, gives rise to higher coolant temperatures, 500°C, which approach those steam conditions found in modern fossil plants. The following description is focused on the superheater sections and their effects on the overall turbine plant performance.

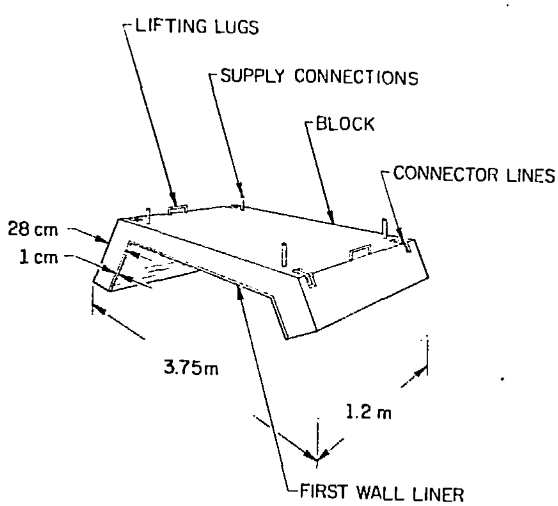


FIGURE 2. Superheater blanket block.

A superheater block is shown in Fig. 2. It represents the more complex geometry of two regions shown in Fig. 1. It is made of three flat plates (blocks) of Inconel 625 joined coolant channel fabrication by welding. The assembled block weighs approximately 5700 kg. The coolant channels, feeder lines, and manifolding passages are formed by gun boring and plug welding the ends shut. Six layers of holes radial to the plasma on 3-cm centers form the grid pattern for the coolant channels. Each channel extends across the block in the poloidal direction. Each layer of tubes is fed by a cross feed channel and all layers are joined by a manifolding channel to which an inlet or outlet supply connection is made (see Fig. 3). All holes are bored to 1.5-cm diameter and filler rods are placed within the bored holes to adjust the coolant velocity (see Fig. 4) with the given mass flow in each of the six channel layers.

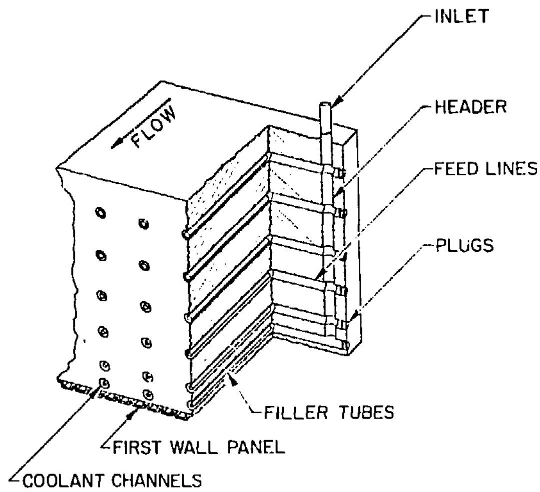


FIGURE 3. Blanket block manifold layout.

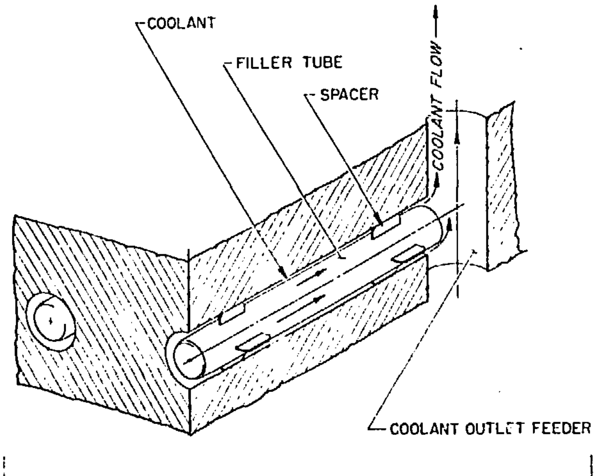


FIGURE 4. Typical coolant channel arrangement.

Orifices are inserted in the channel inlet regions for equal flow distribution after which all open block ends are welded. The superheater block is designed to operate at 8.6 MPa with a central metal temperature of $\leq 650°C$. A corrugated type steam-cooled panel is attached to the front (plasma) face of the block serving as the first wall barrier.

## THERMAL-HYDRAULIC ANALYSIS

The blanket design considered in the analysis is composed of monolithic Inconel 625 blocks, 0.28 m in the radial direction, 1 m in the toroidal direction, and 1.2 m in the poloidal direction. Coolant flows in the poloidal direction through an array of bored channels. Since the heat generation rate decreases exponentially with radial location, the flow area of the coolant channels and their spacing are varied radially to make the temperature distribution in the blanket block as uniform as possible. A set of six coolant channels with iteratively adjusted cross-sectional areas and interchannel distances was evaluated during the first round of analyses (see Fig. 5).

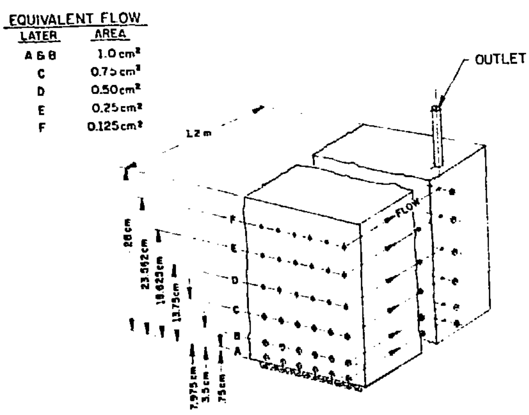


FIGURE 5. Coolant channel model.

A computer code capable of solving a set of three-dimensional thermal hydraulic equations was used to establish the transient and quasi-steady-state temperature distribution within the blanket block. Because of the large thermal inertia of the blanket blocks, the overall thermodynamic efficiency of the power conversion system and the size of the thermal storage system depends significantly on the DT burn cycle. Hence, to investigate the impact of the burn cycle on the performance of ANL/EPR, the following three burn cycles, each within 15-s dwell time, were considered: (1) 65 s; (2) 120 s; and (3) 300 s. The neutron power profile was provided by neutronic calculations for the ANL/EPR.[2] The thermal-hydraulic and power cycle analyses are based on the assumption that the vacuum wall, and the vertical inner and outer blanket is cooled by pressurized water while the horizontal upper and lower blanket serves as the energy source to superheat steam from 300°C to approximately 500°C without exceeding the maximum allowable temperature of 650°C for Inconel 625.[4]

The first step in the thermal hydraulic (TH) analysis is to obtain a set of maximum coolant outlet temperatures within the blanket block. The cross-sectional area of the channels, distribution of coolant channels, and coolant flow rates are based on an iterative set of analyses which results in a series of solid node temperatures that are within the materials temperature limitations. This is the first step in the TH analysis that strives to obtain the maximum coolant outlet temperatures without considering the thermal stresses within the blanket block. The second step is to carry out a stress analysis based on the temperature distribution within the blanket block. If the thermal stresses are too high, then adjustments are made to coolant channel distribution and the analysis repeated. Thus, a final set of coolant channel size and distribution of coolant channels are obtained that leads to least thermal stress and maximum coolant outlet temperature without exceeding the material temperature limitations.

## *STRESS ANALYSIS*

Consideration of the thermal stress distribution in the blanket blocks influences the mechanical design of the blanket in two ways. First, the design of the blanket support structure must permit the overall thermal deformation of the blocks without permitting radial neutron-streaming paths to open between blocks during operation or shutdown of the reactor. Constraint of the overall deformation would result in excessively high stress levels in the blocks. The overall thermal deformation depends on the distribution of heat generation in the blanket and the coolant properties, but it is relatively insensitive to details of the coolant channel arrangement. Second, the temperature distribution within an unconstrained block produces a self-equilibrated stress field having local regions of stress concentration at the coolant channels and hot and cold spots in the block. These stress concentrations depend strongly on coolant channel arrangement; small changes in the channel pattern can correspond to significant alterations of the associated stress levels without affecting the thermal performance of the blanket.

A simple stress analysis procedure was developed to evaluate the relative merits of various coolant channel configurations at the conceptual design level without resorting to the detailed and expensive stress analysis required for a final design. This procedure is based on the separation of the three-dimensional temperature distribution obtained in the thermal-hydraulic analysis described previously into two component distributions which respectively produce the overall deformation effects and the localized stress-concentration effects. A linear (in three-dimensional cartesian coordinates) temperature distribution is fitted to the given temperature distribution by a least-squares technique. The corresponding deformation field is obtained exactly in closed form and represents the global deformation of the block; the associated stress field is determined by interaction with the support structure since a linear temperature distribution produces no self-equilibrated thermal stress field. Then, a residual temperature distribution is computed as the difference between the given distribution and the least-squares fit. An approximate self-equilibrated thermal stress distribution is obtained for this residual problem. The entire procedure has been automated in a small computer program which accepts the output data from the thermal-hydraulic analysis. Comparisons between the relative merits of coolant patterns can thus be obtained rapidly and inexpensively.

## POWER CYCLE ANALYSIS

The combination pressurized water evaporator and direct superheater cycle are shown schematically in Fig. 6. The inner and outer blanket region produces water at 337°C, 15.3 MPa which is delivered to a recirculation-type evaporator via a closed circuit including pumps and pressurizer, with all components similar to those found in a PWR-type light water reactor.

Steam produced in the low pressure side of the evaporator at 300°C, 8.6 MPa, flows into the superheater regions of the blanket and heated to 500°C where it then flows directly into a conventional high pressure turbine. Assuming standard plant conditions, a thermodynamic efficiency approaching 41.5% should be achievable. Thermal storage in the massive Inconel and stainless steel blanket regions reduce the steam temperature variation to a modest 5 to 8°C during the 15-s reactor dwell (plasma exhaust) period. Only a cursory investigation of the plant startup, normal operation, and shutdown has been made, but no acute problems unrelated to tokamak-type reactors is apparent.

## DISCUSSION OF RESULTS

Typical plots (temperature versus distance and temperature versus time) for the three burn cycles are shown in Figs. 7-12. From the thermal-hydraulic analysis, the following observations may be made:

(1) The longer the burn cycles, the higher is the total power output. Although this is not an unexpected result, the peculiar power profile should be pointed out for the fusion reactor during startup and shutdown which leads to a significant power loss.

(2) The longer the burn cycle, the smaller the thermal storage system requirement with the resultant lower overall system cost (assuming a constant dwell period).

(3) The shorter the burn cycle, the longer the quasi-steady-state time.

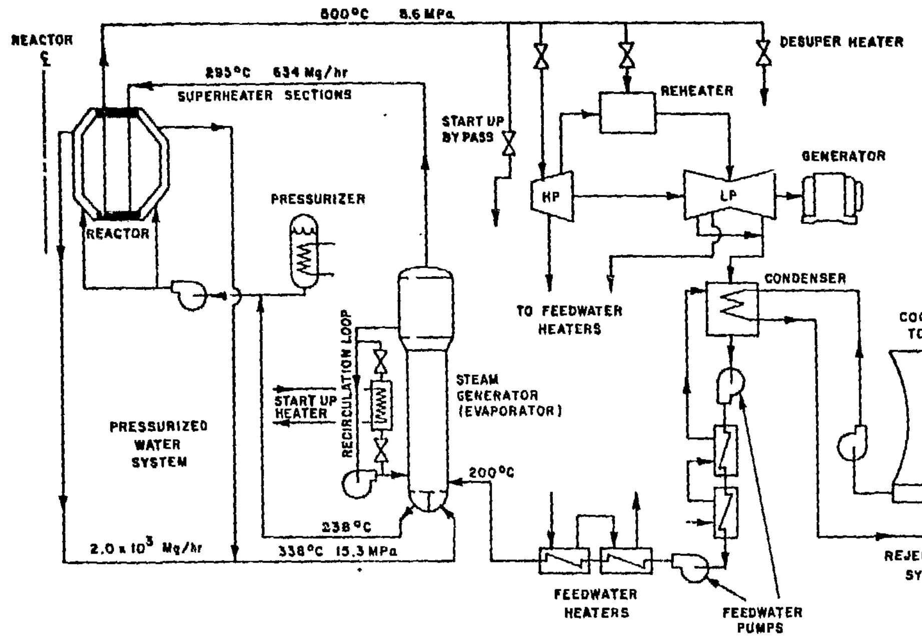


FIGURE 6. EPR dual cycle power system schematic.

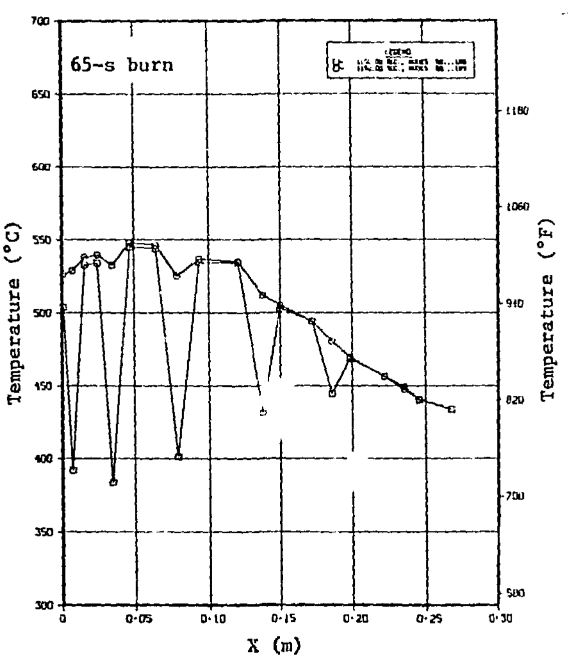


FIGURE 7. Analysis of ANL/EPR blanket, mat-Inconel neutron wall loading - one times normal, tin - 300°C coolant steam, second zone temperature at quasi-steady state.

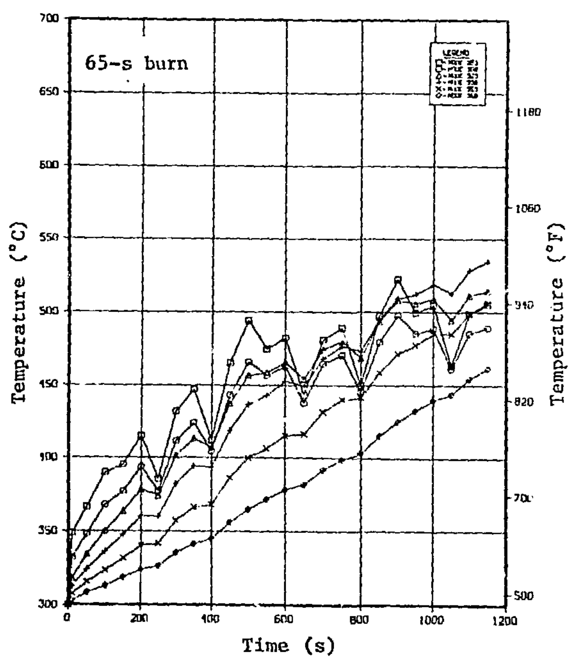


FIGURE 8. Analysis of ANL/EPR blanket, mat-Inconel neutron wall, loading - one times normal, tin - 300°C coolant steam, fourth zone temperature at quasi-steady state.

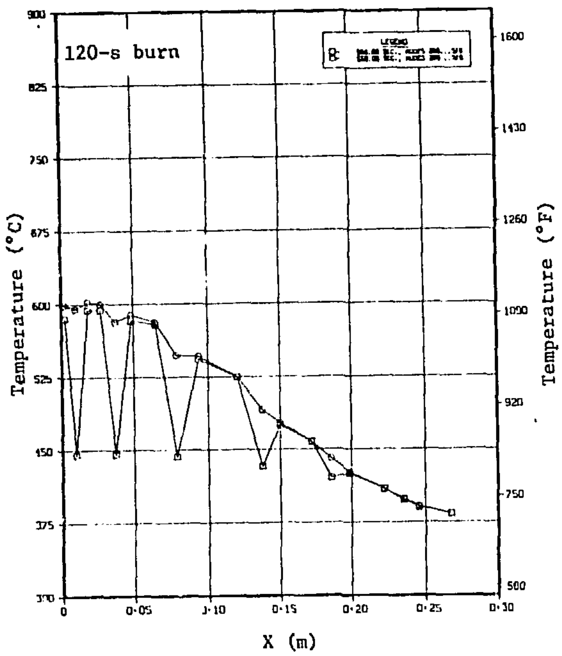


FIGURE 9. Analysis of ANL/ERP blanket, mat-Inconel neutron wall loading - one times normal, tin - 300°C coolant steam, fourth zone temperature at quasi-steady state.

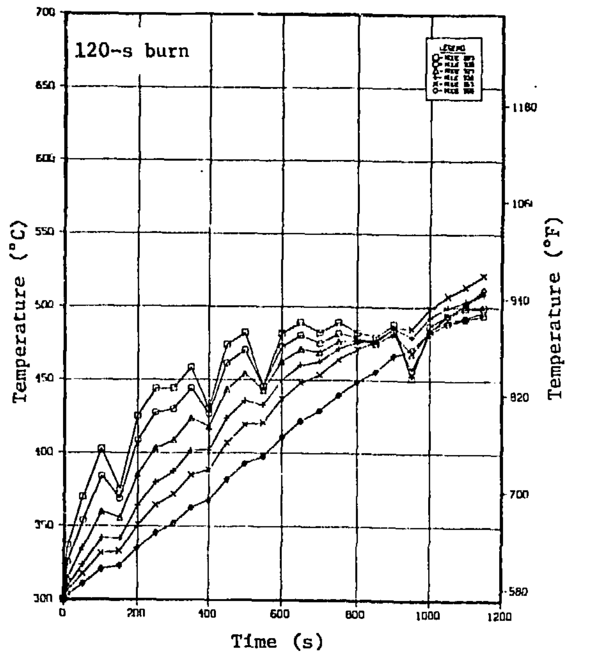


FIGURE 10. Analysis of ANL/EPR blanket, mat-Inconel neutron wall loading - one times normal, tin - 300°C coolant steam, fourth zone temperature at quasi-steady state.

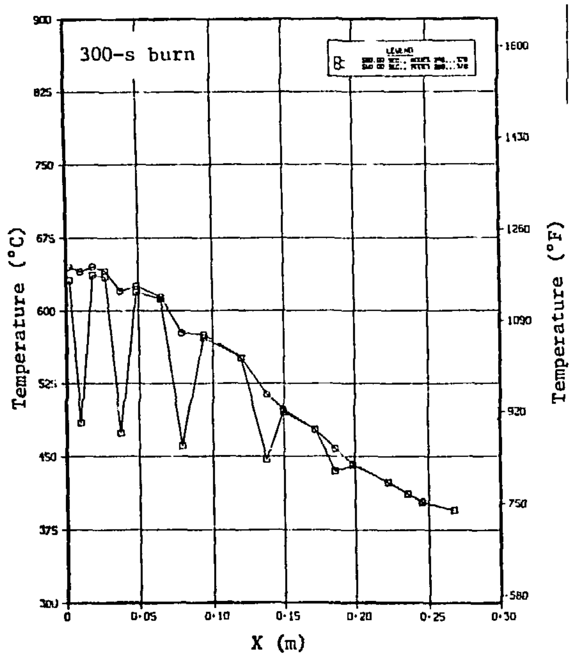


FIGURE 11. Analysis of ANL/EPR blanket, mat-Inconel neutron wall loading - one times normal, tin - 300°C coolant steam, fourth zone temperature at quasi-steady state.

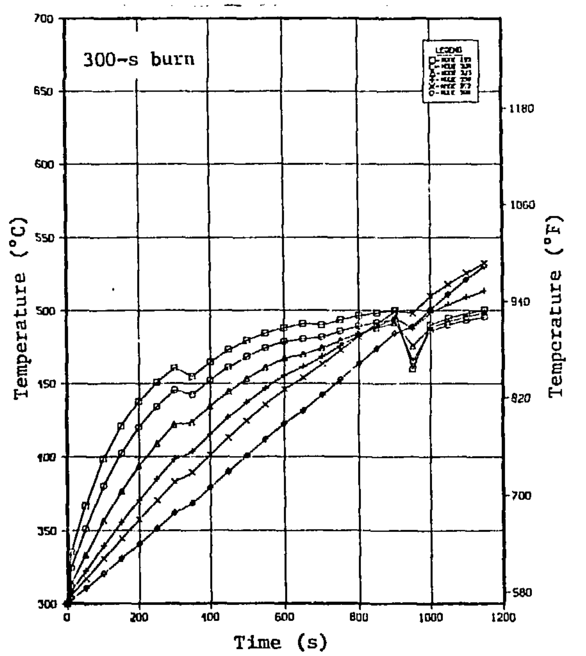


FIGURE 12. Analysis of ANL/EPR blanket, mat-Inconel neutron wall loading - one times normal, tin - 300°C coolant steam, fourth zone temperature at quasi-steady state.

(4) The quasi-steady-state time increases as the channels are removed from the plasma surface. For example, the outlet temperature of the coolant through the first coolant channel approaches essentially quasi-steady state after 600 s. However, the temperature of the coolant through the last coolant channel takes more than 1200 s to approach quasi-steady state (Fig. 8).

Another observation that can be made is that in order to obtain an average outlet temperature of 500°C ($\Delta T$ = 200°C), one can adjust the flow rates such that the outlet temperature for the coolant channels near the plasma is less than 500°C, and the outlet temperature of the coolant away from the plasma is more than 500°C. This has two beneficial effects: (1) it reduces the maximum material temperature near the plasma where the temperature gradients are the highest (high neutron flux); and (b) it increases the coolant outlet temperature away from the plasma where the temperature gradients are not very large (low neutron flux). From an examination of Figs. 7, 9, and 11, it can be seen that even for a burn cycle as long as 300 s, it takes many cycles before quasi-steady-state conditions are approached.

Thermal-stress analyses were performed for the 0.28-m thick Inconel 625 blanket blocks to correspond to the temperature distribution obtained in the thermo-hydraulic analyses discussed previously. Typical thermal strain distributions through the block thickness are shown in Figs. 13-15 for burn cycle periods of 65 s, 120 s, and 300 s, respectively. Strain profiles are shown at form levels measured in the direction of coolant flow. The similarity between the three figures indicate that the strain distribution depends primarily on the channel arrangement and not on the duration of the burn cycle. The relatively large tensile strains at the front of the block near the inlet end and at the back of the block near the outlet indicate that these regions are colder and are less compatible with the temperature and strain distribution found in the center regions of the block. Possible remedies include moving the first and last channels farther from the surface, enlarging the channels in the central region, or increasing the number of channels.

## CONCLUSIONS

The results of the thermal-hydraulic and power cycle analyses show that the thermodynamic efficiency approaching 41.5% can be achieved based on steam as the coolant. The results of mechanical analysis show that preliminary thermal mechanical optimization of blanket block design can be accomplished at the conceptual design level using rela-

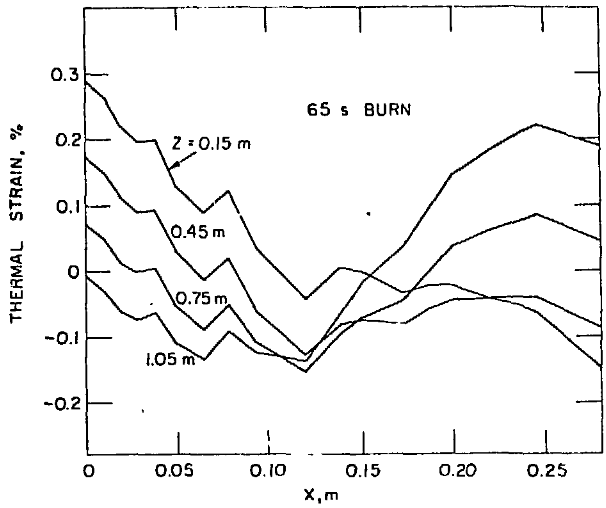


FIGURE 13. Thermal strain distribution.

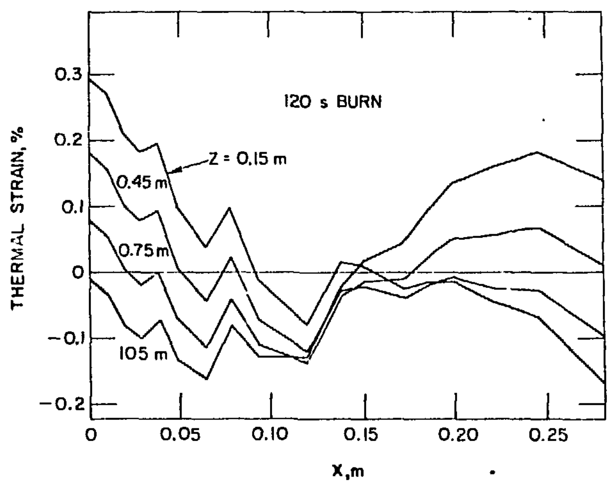


FIGURE 14. Thermal strain distribution.

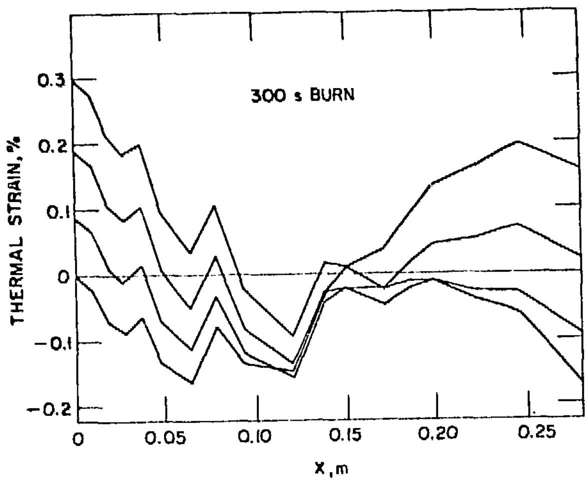


FIGURE 15. Thermal strain distribution.

tively simple computational methods and improvements in strain levels can be accomplished without any significant change in thermal-hydraulic performance. Of course, numerous questions pertaining to tritium containment must be fully addressed before this approach to power cycle operation could truly be considered acceptable. However, from a cursory analysis, a Rankine cycle using steam directly from the blanket system presents no more serious tritium containment problems than a closed loop helium-driven Brayton cycle, which has, in the past, been employed in numerous designs of advanced power conversion systems for commercial fusion power reactors.

## MAINTENANCE

The superheater blocks occupy preferential horizontal blanket positions in the upper and lower regions of the vacuum vessel within which they are supported. This arrangement allows these pieces of blanket along with their attached liner panel to be more easily and quickly removed and replaced than the stainless steel blanket sections. The blanket replacement system used on the ANL/EPR allows for entry into the plasma zone through the upper annular region of the shield without removal of vacuum pumps, neutral beam apparatus, or any major portion of shielding. Removal of a few pieces of upper shield exposes the vacuum vessel cover which upon removal exposes the upper and lower superheater blocks. These are removable with relatively simple apparatus (see Fig. 16). Coolant connections are made through the shield.

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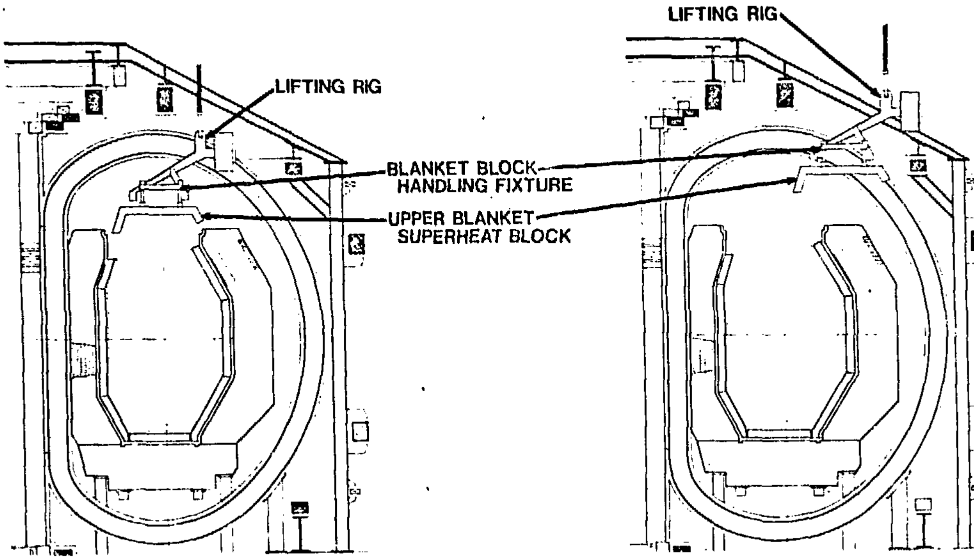


FIGURE 16. Upper blanket block removal.