MECHANICAL DESIGN AND ANALYSIS FOR A EPR FIRST WALL/BLANKET/SHIELD SYSTEM*

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Introduction

Continuing studies are in progress at ANL to expand upon the design of a first wall/blanket/shield FW/B/S system and power conversion for a tokamak type Experimental Power Reactor (EPR). The FW/B/S system has evolved from an earlier design[1] for a low beta, circular cross section plasma (major radius ≈ 6 m) to one for a higher beta elongated plasma with a 4.7 m major radius. Basic mechanical design and layout features of the old and new EPR designs showing some of the more important design developments are depicted in Figure 1 and listed in Table I. These developments are aimed at simplifying the design, reducing the costs and in addition, improving the plant thermal efficiency and overall maintainability. In the area of the reactor blanket, significant thermal hydraulic and stress analysis have been performed to substantiate the integrity of the chosen concept. This paper deals with the discussion of these improved features.

Vacuum Vessel

To improve on the multitude of conditions and interface requirements associated with the vacuum vessel, it first was relocated to the outer portion of the reactor blanket a distance of 35 cm from the plasma boundary. A total of 22 cm of stainless steel blanket and first wall materials intercede to minimize the radiation damage to this component. In this position the vessel as shown in Figs. 1 and 2 serves two functions; namely to provide the required plasma vacuum

Table I. Comparison of EPR-77[2] and EPR-76 Design Features

Reactor Parameters	EPR-77	EPR-76
Average toroidal beta, β_t	0.08	0.048
Major radius, R (m)	4.70	6.25
Plasma minor radius, a (m)	1.34	2.10
Plasma elongation, b/a	1.65	1.0
Aspect ratio, A	3.5	3.0
Vacuum chamber volume, V_c (m^3)	450	711
Blanket thickness, Δ_B (m)	0.28	0.28
Shield thickness		
inside, Δ_s^i (m)	0.52	0.58
outside, Δ_s^o (m)	0.97	0.97
TF coil bore (m)	6.8 x 10.0	7.8 x 12.6

Design Features	EPR-77	EPR-76
Vacuum Vessel	Outer blanket location, heavy 7-10 cm wall	Inner blanket location, thin 2 cm wall
First Wall Panels	Flat sections structurally attached to blanket	Curved sections separately replaced
Blanket	Slab type blocks, 128 major pieces	Contoured blocks, 272 pieces
Shield	Flat surfaced blocks, 192 pieces	Curved surface blocks, 420 pieces
Maintainability	Single purpose large reactor top closure	Vertical vacuum limited horizontal ports
Power Conversion	Dual cycle system eff. ∿ 39%	Pressurized water 25-30%

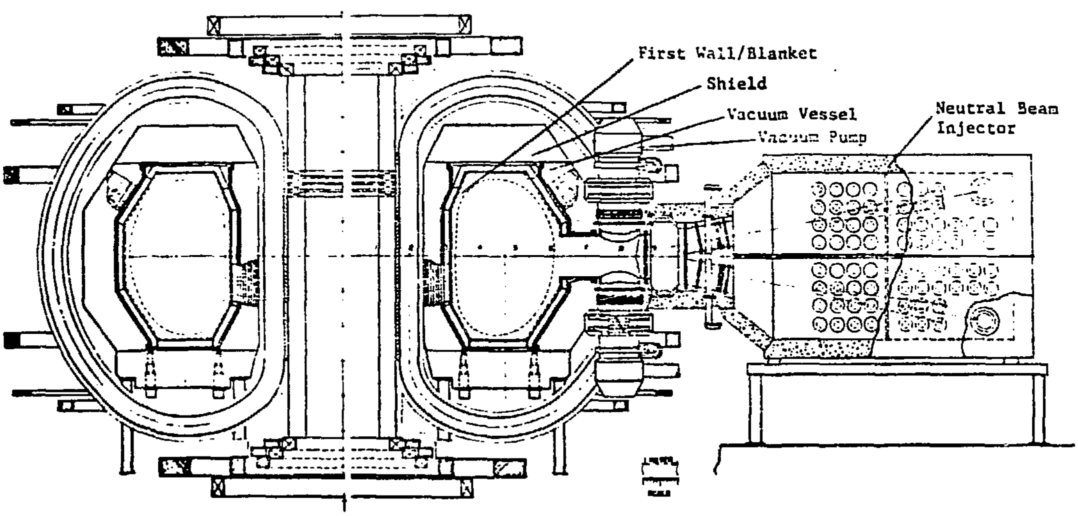

Figure 1. ANL Tokamak Experimental Power Reactor.

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conditions, and secondly to support the blanket and first wall panels. It is also the temperature interface boundary between the hot fluids of the energy conversion system and the much cooler reactor shield system.

The vessel is made of flat plates assembled in three basic pieces per section with sixteen sections forming the irregular toroidal chamber whose overall dimensions are 13 meters in diameter by 5.4 meters high. The vessel weighs ~ 260 Mg and has 12 required horizontal beam ports. The vessel plates made from Type 316 stainless steel are nominally 7 to 10 cm thick which are formed by welding into the 3 pieces shown in Fig. 2 forming a section. The sixteen sections are bolted together (see Fig. 3) using a joining bar section and appropriate bolting from the inside of the vessel after which an expansion section covers the bolting seam with a welded seal strip. The top closure is also bolted and seal welded and functions solely for repair and maintenance access of the vessel internals. Each vessel plate contains a series of parallel gun bored coolant channels poloidally oriented in respect to the reactor plasma. Coolant in each plate is distributed through these channels in internally bored headers which are manifolded and carried to upper and lower supply and return rings. Support for the vessel is achieved with four adjustable column units located at the bottom of each segment which are in turn anchored to the lower shield block.

Some of the advantages this design offers are: relative ease of fabrication; modest radiation damage effects; modest cooling problems; accommodation of thermal expansion and insulation; ease of access to internals with a large area single purpose closure; repairability and replaceability with relative ease.

First Wall/Blanket

The first wall panels are integrated with the blanket blocks and line the interior of the vacuum vessel as shown in Fig. 1 following the basic contour thereof. These blocks, 128 in number are supported through the bottom of the vacuum vessel and anchored to its sides. Coolant line connections from each block are made through the top and bottom of the vessel where there is access in a relatively uncongested area. All blocks are readily removed through the vessel closure with special handing fixtures.

The blanket block arrangement is depicted in Fig. 4. It consists of 3 inner blocks, 3 outer blocks a bottom and a "U" shaped upper section. Each block is made from a 20 cm thick Type 316 stainless steel flat plate, with a coolant channel arrangement as shown. The coolant passages are transverse to the reactor and graded to match the energy distribution within the block thickness.

The upper and lower blocks are specially designed for steam cooling. Thermal and stress analysis indicate that steam exiting 410°C at 8.2 MPa is acceptable while the remaining inner and outer blocks operate with pressurized water cooling at 340°C and 152 MPa. This combination of coolant using superheated steam in direct cycle has a 39% thermal efficiency potential.

• Thermal Hydraulic Analysis -- The ANL/EPR blanket system utilizes an assembly of modular stainless steel blocks with gun-bored coolant channels. The thermal hydraulic analysis is based on a single modular block that is 0.25-m thick and 1.5-m in length in a plane parallel to the first wall. To simplify the geometric model, co-radial groups of coolant channels

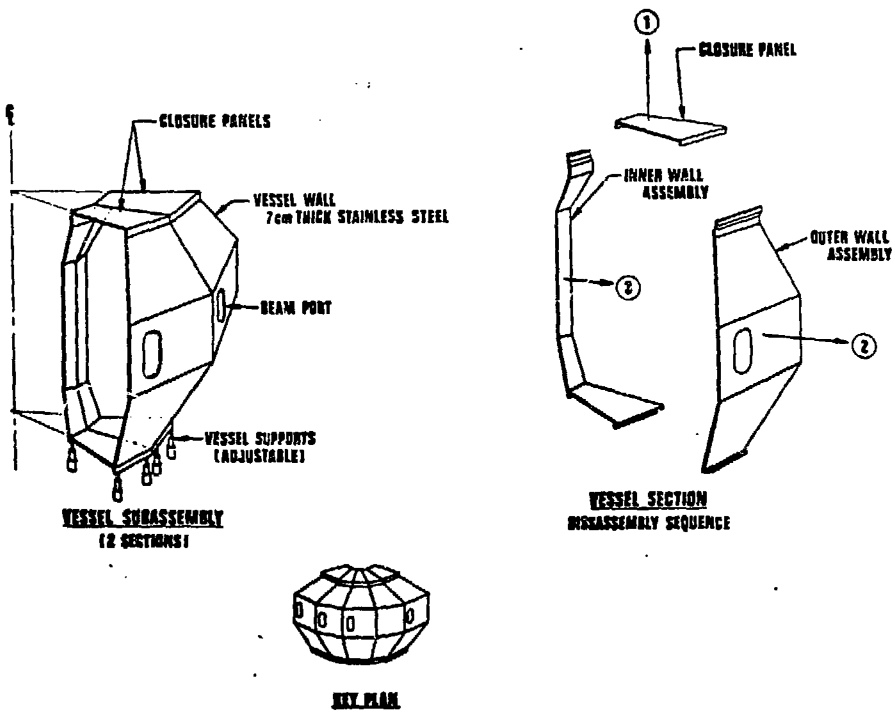

Figure 2.

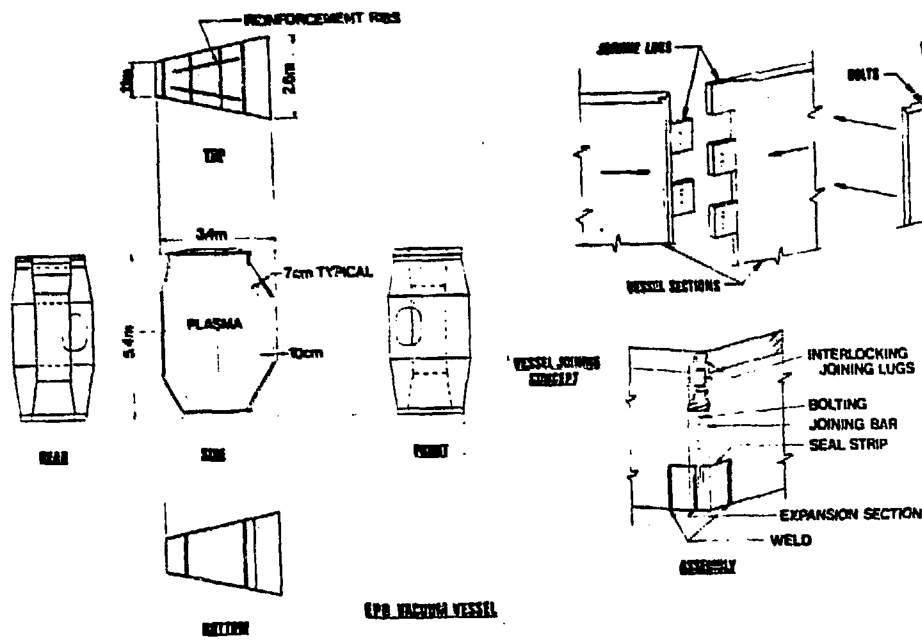

Figure 3.

with graded cross-sectional areas (to compensate for the exponential attenuation of neutrons in the blanket block) were selected for the initial analysis. 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. The neutron power profile was provided by neutronic calculations for the ANL/EPR based on a 65-s burn pulse followed by a 15-s dwell period between pulse.

The operating conditions for the three coolant systems are summarized in Table 2. These operating conditions are consistent with those for existing pressurized water reactors, gas cooled reactors and fossil powered steam plants. Also, one of the salient features of this analysis is that is includes low pressure steam as a coolant and direct cycle working fluid for a fusion power plant.

A set of six coolant channels with iteratively adjusted cross-sectional areas and interchannel distances (see Figure 4) was evaluated during the first round of analyses. The analytical results are included in Table 2. Because of significant differences in the physical and transport properties of the three coolant systems, the coolant velocities, Reynolds numbers and heat transfer coefficients are seen to vary over a wide range. The lower heat transfer coefficient for steam results in a need for shorter interchannel spacing in the direction normal to the first wall. Hence, for a given reactor, cooling with steam appears to require more coolant channels per unit volume of blanket than either water or helium. Since the length of the coolant channels is only 1.5 m, the frictional pressure drop across the channel turns out to be negligible for all cases. This indicates that one can minimize the large variations in required fluid velocity by judicious selection of coolant channel cross sectional

Table 2. Summary of Thermal Hydraulic and Power Cycle Analysis

Parameter/Coolant	Water	Helium	Steam
Pressure, MPa	15.17	5.17	4.83
Interchannel Distance, cm	5	5	3
Inlet Temperature, °C	240	205	260
Outlet Temperature, °C	340	400	410
Maximum Blanket Temperature, °C	500	550	550
Velocity, m/s	0.06–0.18	12–26	2–5
Reynolds Number x 10^{-3}	3–13	10–30	12–50
Heat Transfer Coefficient, W/m^2-K	1200–2550	1100–1900	400–700
Pressure Drop, MPa	< 0.01	< 0.01	< 0.01
Thermodynamic Cycle Efficiency, %			
Direct Cycle	—	13	39
Indirect Cycle	34	36	36

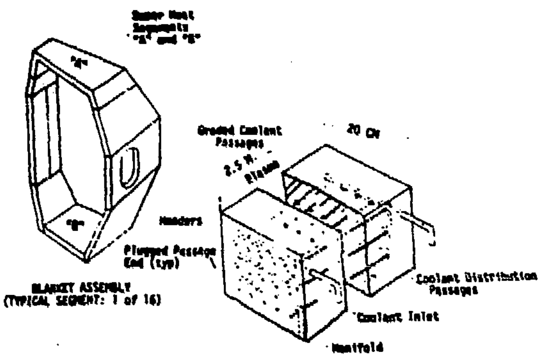

Figure 4.

area and can thereby optimize the heat transfer characteristics.

• **Blanket Stress Analysis —** Consideration of the thermal stress distributions in the blanket blocks influences the mechanical design of the blanket in two significant 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 cartesion 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.

As an example, thermal-hydraulic analysis using pressurized water coolant indicated that channels spaced 5 cm apart in the toroidal direction and located at radial positions of 1, 4, 8.9, 13.8, 18.8, and 23.8 cm from the inner surface would provide a reasonably uniform temperature distribution in the block; the channels varied in cross-sectional area between 1 cm^2 and 0.5 cm^2 (see Fig. 5). The associated maximum thermal strains in the block were found by the approximate procedure described above to be 0.35%. Moving the first channel to 1.25 cm from the surface reduced the strain to 0.25%. Moving the first channel to 1.5 cm and the second channel to 4.5 cm further reduced the maximum strain to 0.18%. The improvements in strain levels were accomplished without any significant change in thermal-hydraulic performance. The results indicate that preliminary thermal-mechanical optimization of blanket block design can be accomplished at the conceptual design level using relatively simple computational methods.

The overall unconstrained thermal deformation is 0.6% of the room-temperature dimensions, and the blocks become convex toward the plasma. These overall dimensional changes are virtually independent of the channel pattern variations discussed above. Stress analyses performed for temperature distributions at several times during the burn cycle indicate that the large thermal inertia of the blanket prevents any significant cyclic strain variations.

• **First Wall —** The first wall panels are shown in Fig. 6, they are made from 316 type stainless steel and form the same contour as the mating blanket block to which they are attached. The panels are similar to those described in the earlier ANL EPR design,[1] having a close array of transverse coolant channels connected to integral return and supply headers which are fed from lines common to the mated blanket block. The plasma face of the panel will be coated with a low-Z material. The blanket blocks are arranged in the reactor in stacks three high on the inner and outer walls of the vessel. At each segment junction a small 10 cm wide segment filler block (Fig. 7) is used for "packing" the block system. Each small block, 96 in all, is similar in construction to the first wall/blanket units with a first wall panel section integrated into a blanket block. The first wall panels in this case will have additional cooling capacity and may serve as plasma limiters. These corner blocks are removed first, freeing the sides of the major block to be removed. Since the major blocks are stacked, it may require removal of eight pieces to gain access to a lower vertical blanket block.

Some of the advantages of this first wall/blanket design are: simple and inexpensive construction of blanket blocks and first wall panels; small numbers of components and fewer coolant connections; superheater blocks for high thermal plant efficiency; relatively simple and quick maintenance and replacement function; experimental testing of superheater modules or large portions of the first wall with relative ease; accommodation of eddy currents effects.

Shielding

The shield is divided into an inner and outer block denoting a high efficiency B_4C-SS composition for the inner shield and less expensive SS-graphite-boron-lead mortar-aluminum composition in the outer shield. There are 192 blocks that envelope the vacuum vessel (see Fig. 8). The configuration is

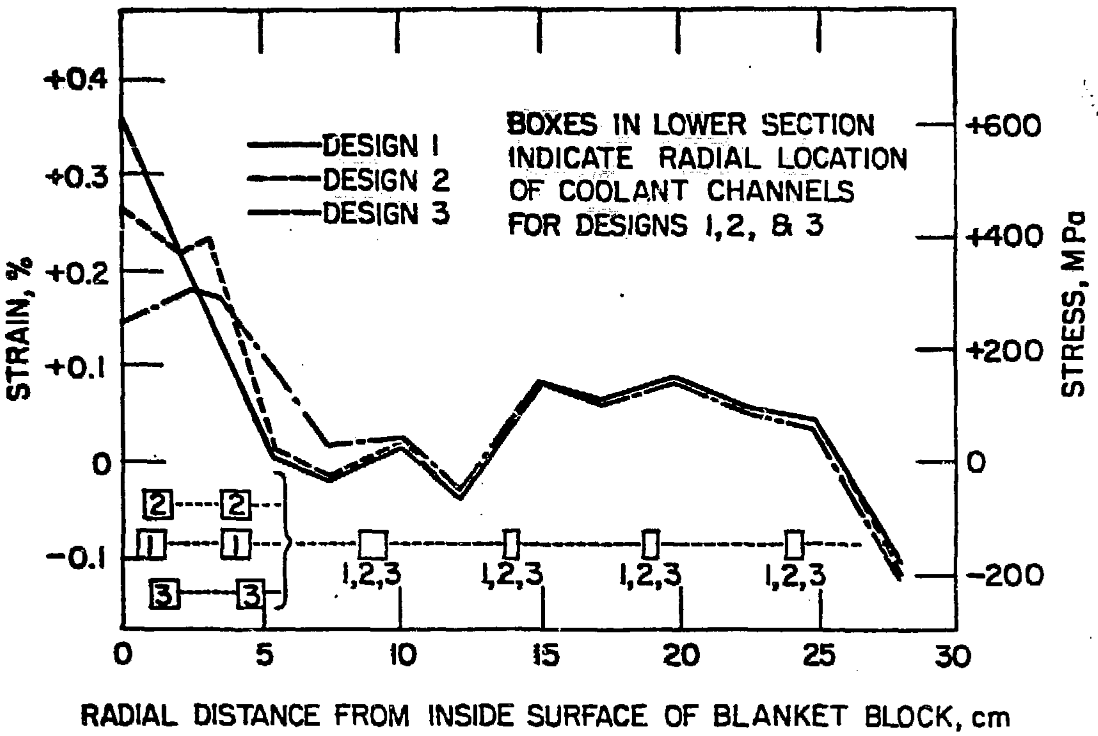

Figure 5.

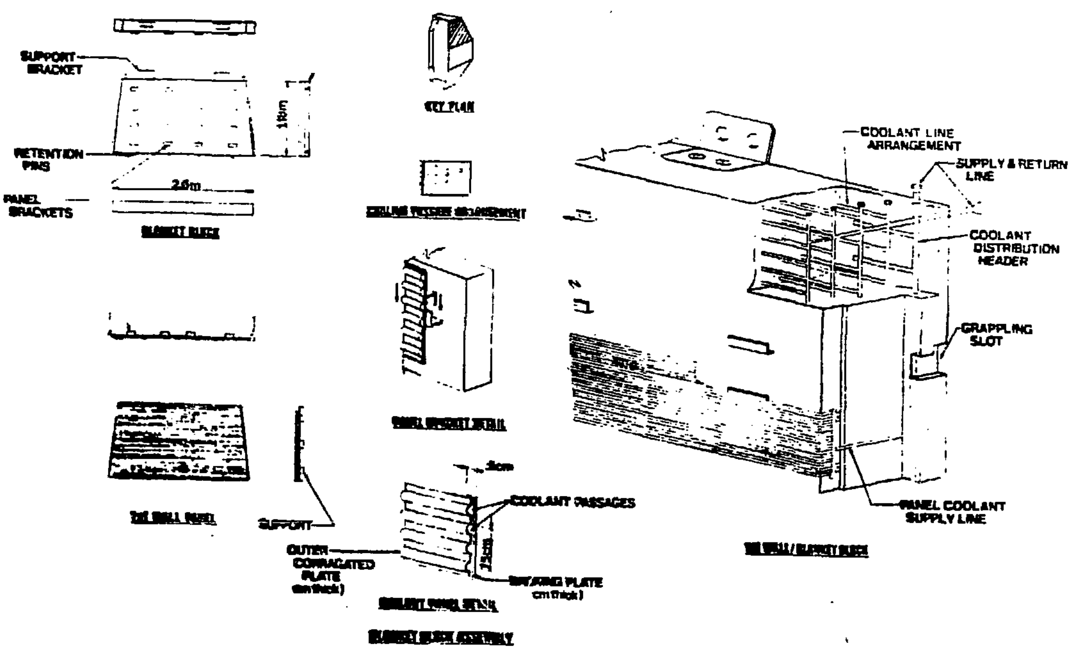

Figure 6.

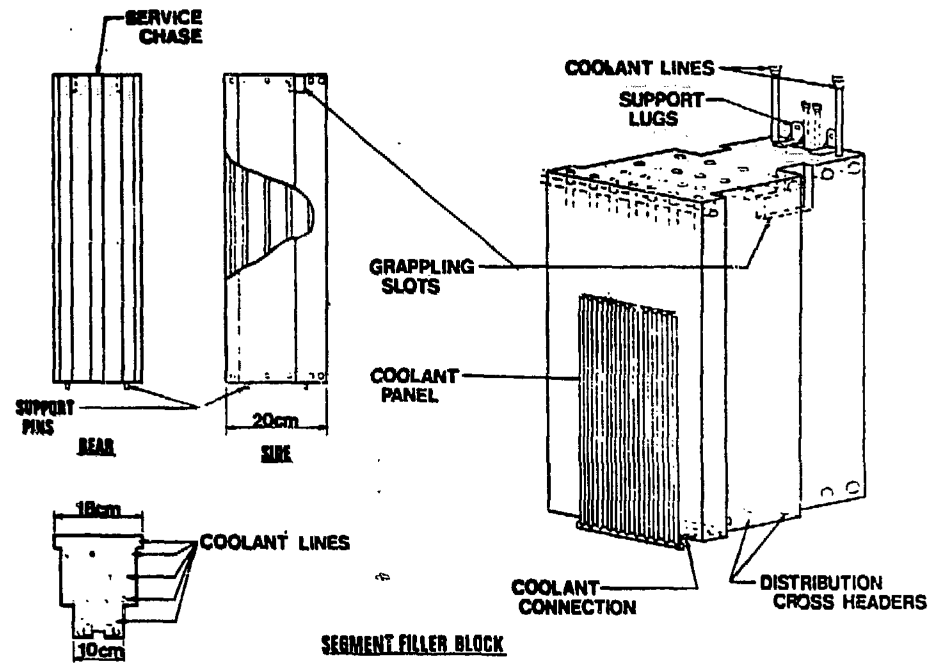

Figure 7.

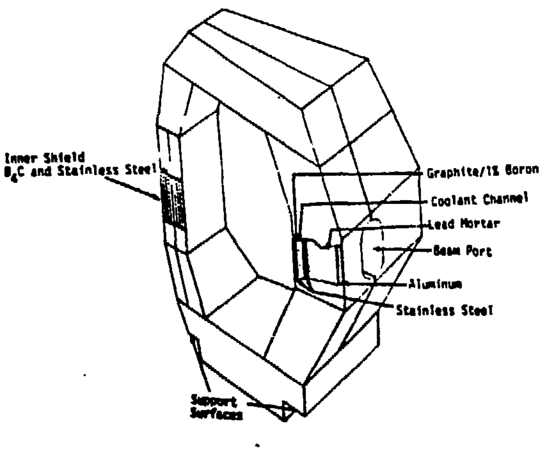

Figure 8.

designed to provide adequate shielding in a minimum number of easily fabricated pieces with special emphasis on simple and rapid access to the vessel closure. The shield is supported on columns from beneath the reactor and in turn supports the vacuum vessel/blanket/first wall complex. A low pressure water system cools the shielding blocks. The advantages of this design are: ease of manufacture (small numbers of blocks with flat surface); ease of support and handling; quick and simple access to the reactor vessel; favorable magnetic permeability characteristics.

Maintainability

Of prime importance in this reactor design was to obtain simple and quick access to the interior of the reactor vessel for inspection, maintenance repair to obtain the maximum data and availability on this first round power reactor. In addition, experimental modification and total replacement and repair of all major components should be possible with reasonable pieces of remote handling equipment performing in a reasonable time span. In this regard the design of this reactor accomplishes a number of important functions towards satisfying these goals.

- **Vessel Access** — Access to the interior of the vacuum vessel has been simplified over earlier concepts. A single purpose large area closure for access to the vacuum vessel has been incorporated whose sole function is inspection, maintenance, and repair of internal reactor components. This replaces the dual purpose limited area vacuum/experimental port access used previously.[1] The current design has the advantage that access is gained in any given zone by the removal of 4 wedge shaped top shield blocks without disturbing any of the reactor components. A simple lifting fixture is used with hands on operations until actual removal of the block (see Fig. 9). This feature saves considerable time and reduces the risk of damage as in the case of dual purpose ports.

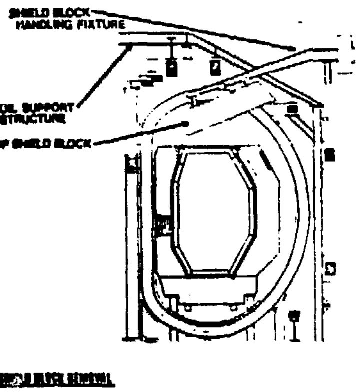

Figure 9.

Once access to the vessel has been made adequate space and provisions are available for uncoupling coolant lines and instrumentation through the vacuum barrier to the blanket and first wall components. In addition the reactor closure flange wall provides a very substantial base from which to mount and index inspection and handling fixtures as shown in Fig. 10. Special consideration has been given to handling of the upper and lower superheater blanket blocks since these may require attention at an earlier interval and because they represent an experimental volume where examination and testing may be easily carried out.

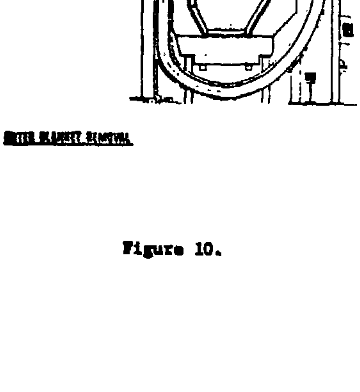

Figure 10.

• Beam Ports — In simplifying the reactor layout the number of major reactor service ports has been greatly reduced, from 48 to 16 in the current design of which only 12 are committed to normal operation. By the arrangement shown in Fig. 1 each port serves two functions as a vacuum exhaust port and as a neutral beam injection port. Two cryosorption pumps and 2 neutral beam lines use a single port complimentarily with each component having its own independent isolation and shield valve. All units are easily maintained and under most circumstances failure of a single component will not lead to a reactor outage. It is the compactness of this vacuum/neutral beam port arrangement that allows the vacuum vessel access system concept to be achievable.

• Secondary Vacuum System — A secondary vacuum system whose boundary envelops the entire reactor shield at the TF coil periphery has also been incorporated in the design of the reactor. The purpose of this feature is to provide a backup system for the primary reactor vacuum systems such that atmospheric contamination of the plasma due to single small vessel leaks is not possible, and secondly; as a safety feature to provide a secondary barrier for detection and entrapment of tritium. These functions serve to minimize reactor outages due to small defects, warn operations personnel of impending leak problems, provides monitoring for tritium contamination prior to handling reactor components, and introduces a reasonable volume in which to promptly handle a massive tritium release.

The secondary vacuum containment which has been called the "bell jar approach" in this design is formed by adding additional vacuum cover plates to the existing torque shell used to restrain the TF coil system designed for ANL by the McDonnell Douglas Corporation.[3] Since the location of these panels is in a very low radiation field, seals may be made with neoprine gaskets using "hands on" maintenance operations.

• Major Components — Maintenance of major components has been simplified to a great extent. In particular by the ability to replace the entire or any part of the vacuum vessel under reasonable conditions in a relatively short schedule insofar as reactor component dissassembly, manipulation and storage is concerned. Adequate space has been provided in the reactor building for complete laydown and storage of reactor components. It is also possible to remove a TF coil by local dissassembly, removing only portions of the adjacent vessel and inner and outer shielding blocks. Replacement of this component with this degree of modest difficulty is an asset to the early generation of power reactors.

Hot cells have been provided for the more routine functions of examination, repair and replacement of first wall panels, blanket, and superheater blocks. These cells also serve the needs of the experimental program.

Power Conversion System

One outstanding development in the EPR design has been the upgrading of the plant power conversion efficiency from $\sim$ 30% to a potential 39%. This was accomplished by using the upper and lower blanket as superheater sections while the inner and outer vertical blanket zones are cooled with pressurized water. To accomplish this, the blanket blocks in the superheater region require a greater number of and different sized coolant channels which were determined by the aforementioned thermal hydraulic analysis. The overall thermal expansion of the blocks will differ and the superheater section will be at a higher overall temperature, thus the geometrical arrangement of the blanket assembly was designed to accommodate these differences by placing the superheat zones in the upper and lower regions of the vacuum vessel. In these extremities additional shielding was added to compensate for voids and streaming gaps.

Two distinct cooling systems are used; the pressurized water system is a closed cycle system consisting of a circulating pump and steam generator identical to that of the current commercial Pressurized Water Reactor (PWRs); and a direct cycle superheated steam system wherein the steam goes directly from the reactor to turbine similar to those used in Boiling Water Reactors (BWR). The dual cycle system is shown in Fig. 11. Pressurized water from the first wall/blanket assemblies of the inner and outer blanket delivers thermal energy to an evaporator operating at 13.8 MPa (2000 psi). The steam thus generated in the evaporator is fed into the upper and lower superheater blocks where it is heated to 410°C at ~ 8.6 MPa (1250 psi) and fed directly into the turbogenerator with a resultant overall cycle efficiency of ~ 39%. Radioactive transport problems of the direct cycle are being analyzed, but they do not appear to be appreciably more severe than those encountered in BWRs. The power plant rating for the EPR is 270 MWth, 120 MWe with a full capacity cooling tower 315 MWth. Cyclic temperature effects on the coolant leaving the reactor are minimized < 2°C due to the large thermal inertia of the stainless steel blanket.

Further analysis using higher strength materials and configurations shows promise for higher cycle efficiencies in excess of 40% where use of modern turbines vs the saturated units now employed in the Light Water Reactor nuclear plants.

References

1. W. M. Stacey, Jr., et al., "Tokamak Experimental Power Reactor Conceptual Design," Argonne National Laboratory, ANL/CTR-76-3 (August, 1976).

2. W. M. Stacey, Jr., "Tokamak Experimental Power Reactor," Argonne National Laboratory, ANL/FPP/TM-93 (November, 1977).

3. C. A. Trachsel, "Development and Evaluation of TEPR Support Structure Concepts," McDonnell Douglas Astronautics Co., East, Contract 31-109-38-3391, ANL (May, 1976).

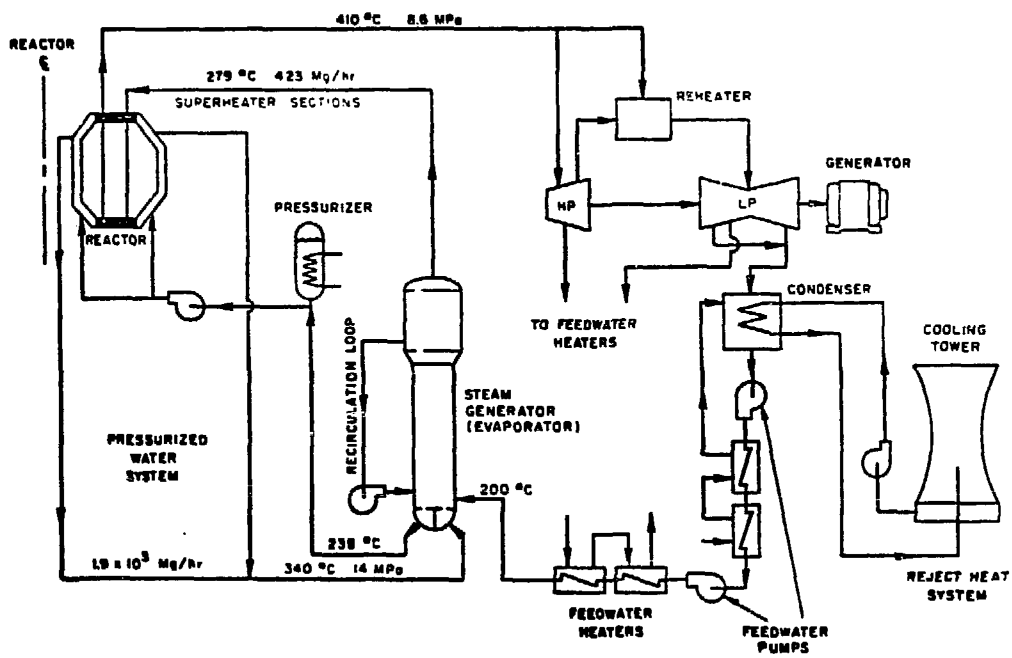

Figure 11.