II. EXPERIMENTAL POWER REACTOR

A. Energy Storage and Transfer Program

R. Fuja, R. L. Kustom, R. P. Smith, R. Wehrle, EPR Project

The energy storage and transfer systems that have been proposed for TEPR[1] and described in the previous FPP Quarterly Report are under experimental assessment and construction.

An inductor-convertor (IC) bridge[2] was built along with two superconducting solenoid magnets for modeling of the EF coil power supply. Tests on this system show experimental results agree closely with predicted theory. Measurements indicate that at least 95% of the energy, in the storage coil at the start of the transfer cycle, transferred to the load coil. The energy transfer process is reversible and the energy in the load coil has been returned to the storage coil. Polaroid pictures (Figure II-1) show the load voltage waveforms at various times in the transfer cycle.

Studies for the control of energy transfer are continuing with the objectives of developing a digital controller that will allow for the transfer and return of energy at a controlled and variable rate.

Algorithms which predict the unbalances which occur when the energy transfer rate is varied, and a method for correcting these unbalances has been developed. Work is starting on the hardware implementation for these algorithms. Protective circuits for IC bridge failures are also being studied. Circuit design to automatically make up the transfer losses will be studied and a digital controller incorporating all of these ideas will be developed.

A model homopolar generator (Hope 1), used for energy reversal in the OH coil of TEPR is under construction. Mapping of the magnetic field between the pole tips has been completed and the data reveals unbalances in the magnetic field along the ϕ and Z axis, of a cylindrical coordinate system. Methods for the elimination of these irregularities have been proposed. They include selective shimming of the pole tips and the installation of corrective coils near the pole tips. Computer studies and preparation for experimental evaluation of these ideas are under way.

The construction of the cylindrical energy storage rotor, with its air support bearings and brush support structure has been completed and mounted to the magnet's central support yoke. Optical circuits for measuring rotor growth in the radial direction are being assembled. When storing 40 kJ of energy and rotating at 2340 radians per second the radial growth of the rotor is expected to be 0.010 inches or approximately 2.54×10^{-4} meters. The effects of this growth on the operation of the air bearings will also be measured and studied.

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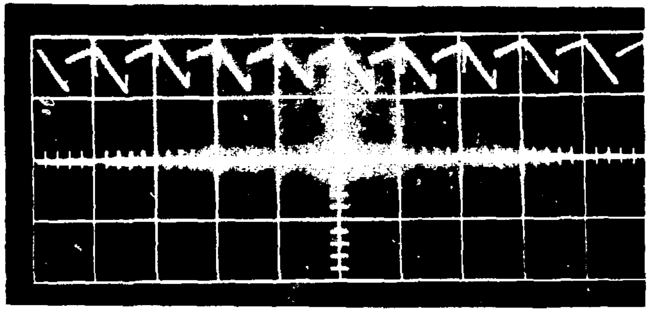
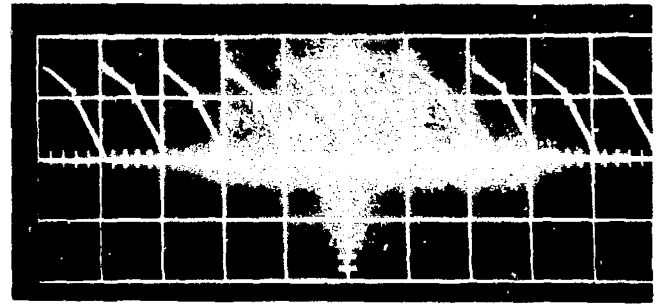
[1] W. M. Stacey, Jr., et al., "Tokamak Experimental Power Reactor Conceptual Design", ANL/CTR-76-3 (1976).

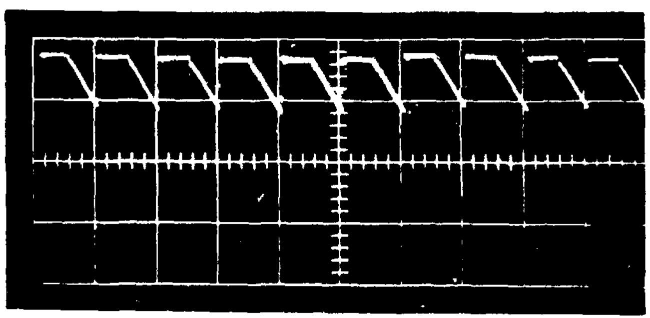
[2] R. L. Kustom, et al., "The Use of Multiphase Inductor-Convertor Bridges as Actively Controlled Power Supplies for Tokamak EF Coils," ANL/FPP/TM-78, (1976).



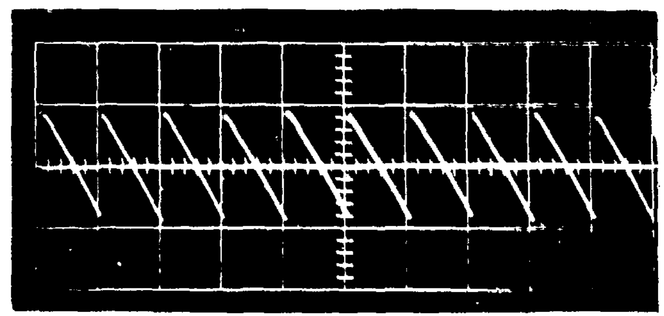
(1) 9% Energy Transfer - 0.5 s after start.



(3) 50% Energy Transfer - 2.0 s after start.



(2) 30% Energy Transfer - 1.0 s after start.



(4) 95% Energy Transfer - 3.3 s after start.

Figure II-1. Load Coil Voltages - 100 v/cm 0.2 ms/cm, Phase Shift 90 Degrees Leading.

Switching circuits which will connect Hope 1 to its load, a 13.6 mh inductor representing the OH coil for TEPR, have been designed, constructed and installed.

Low field, low voltage operation of Hope 1 will begin shortly. However, high voltage operation will have to wait until the field irregularities are minimized.

B. Thermomechanical Analyses

B. Misra and C. K. Youngdahl, EPR Project

The current design concept for the ANL/EPR blanket system consists of 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 0.28-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 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. A set of six coolant channels with iteratively adjusted cross-sectional areas was evaluated during the first round of analyses. 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 pulses.

The results obtained for three different coolant fluids (i.e. pressurized water, helium, and steam) are summarized in Table II-1. The operating conditions indicated in the table are consistent with those for existing pressurized water reactors, gas cooled reactors and fossil powered steam plants. Because of significant differences in the physical and transport properties of the three coolant systems, the coolant velocities, Reynolds numbers and heat transfer coefficient for steam result in a need for shorter inter-channel 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 area and can thereby optimize the heat transfer characteristics.

From the standpoint of maximum thermodynamic efficiency, it seems most advantageous to use a direct Rankine cycle with steam and a closed loop Brayton cycle with helium. However, considerations of tritium containment in direct cycle systems may ultimately force the use of intermediate heat exchangers for the near-term power plants, thus reducing the achievable power conversion efficiency. In order to estimate the magnitude of this reduction, efficiencies were calculated for steam and helium with and without intermediate heat exchangers. Significant improvement in efficiency for steam cooling can be achieved with the direct cycle approach (see Table II-1).

Table II-1. Summary of Thermal Hydraulic and Power Cycle Analyses

Structural Material: Stainless Steel

Neutron Wall Loading: 0.62 MW/m^2

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, $W/m^2 \cdot K$	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

The very low efficiency indicated in Table II-1 for direct cycle helium is a result of the fact that the 400°C exit temperature is far below the threshold temperature for useful operation of helium turbines. From this study, it may be concluded that all three coolants can be used for near-term power producing fusion devices. Since direct cycle steam appears to offer somewhat higher efficiencies than water and helium in temperature limited systems, it merits more serious consideration as a potential power cycle concept for fusion power plants.

Mechanical response analyses for the blanket blocks shown in Figure II-2 were carried out based on the temperature distributions obtained from the above described thermal hydraulic analyses. The resulting thermal stress distribution has local regions of stress concentration at the coolant channels and at other especially hot or cold spots in the block. These stress concentrations are highly sensitive to channel arrangement; small changes in the channel pattern can result in 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

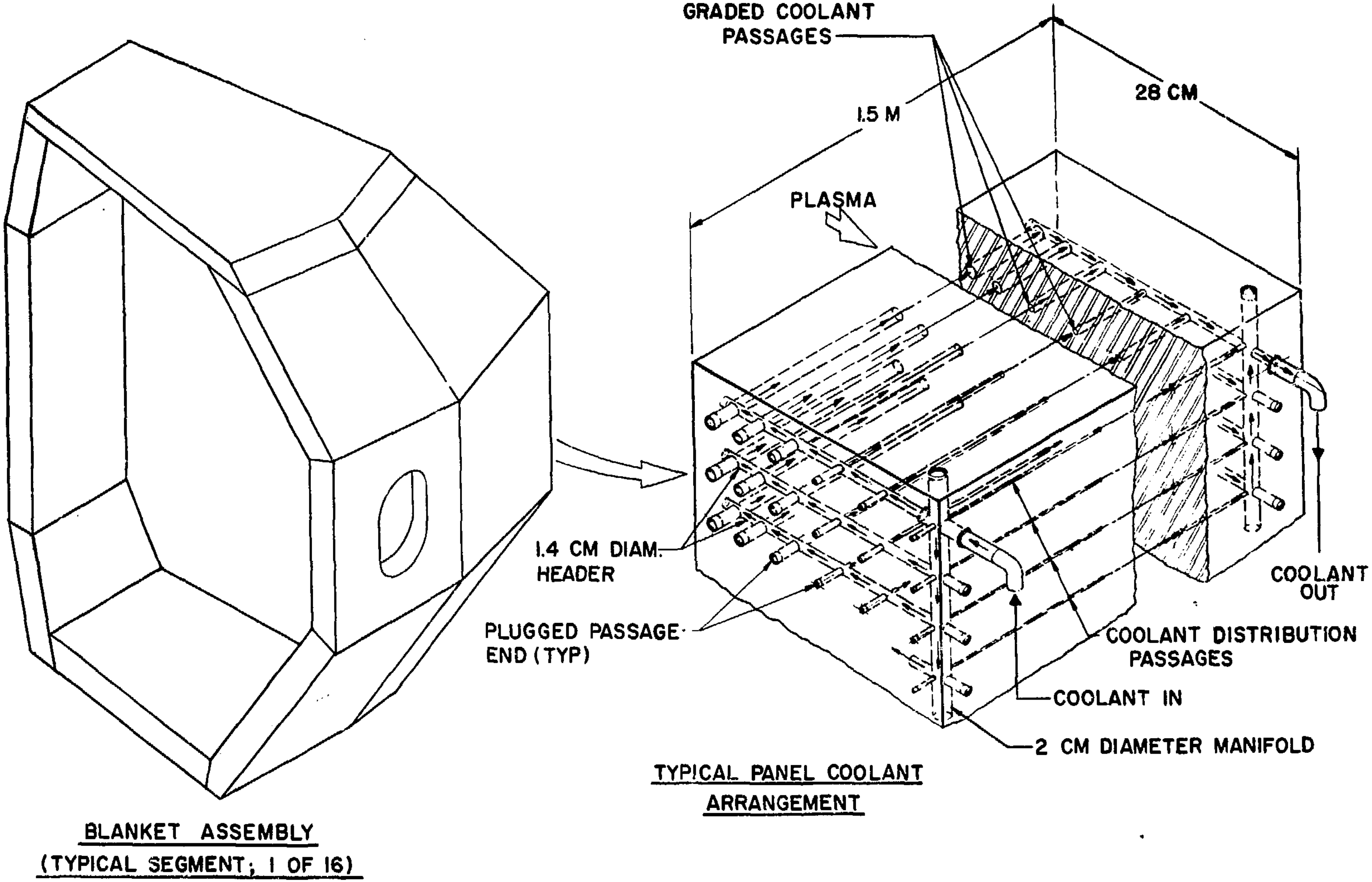


Figure II-2. Coolant Channel Arrangement for Stainless Steel Blanket Blocks Cooled with Water, Helium or Steam.

level without resorting to the detailed and expensive stress analysis required for a final design. The solution to this procedure is based on 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 channel patterns can thus be obtained rapidly and inexpensively.

As an example, thermal-hydraulic analysis using pressurized water coolant (see Figure II-3) 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. 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.

C. A Steam Dual-Cycle Power Conversion System for the EPR

H. Stevens, EPR Project

Based on the results of the thermal hydraulic analysis of the EPR blanket[3] and the associated structural analysis (see Section B above). A dual cycle superheated steam system is proposed for EPR power conversion. This system which utilizes reactor superheating blanket modules delivers superheated steam directly to the turbine with a potential gross conversion efficiency of 39%.

[3] W. M. Stacey, Jr., et al., Fusion Power Program Quarterly Progress Report, January-March, 1977, ANL/FPP-77-1, (1977).

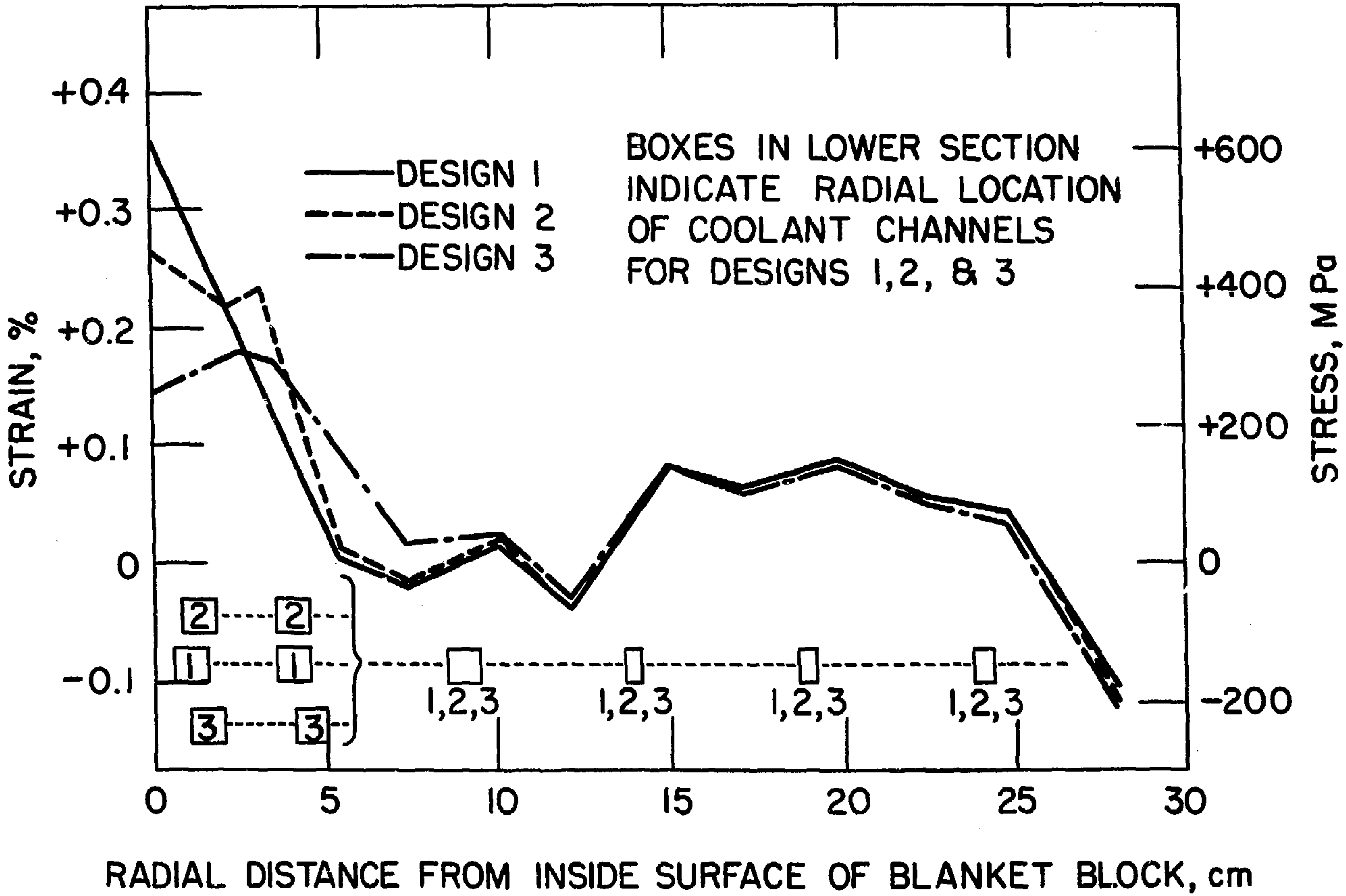


Figure II-3. Thermal Strain Distribution in Radial Direction for Various Channel Arrangements.

The dual cycle is made up of an indirect pressurized water system coupled with a direct cycle superheat system. The pressurized water is used to deliver heat for evaporation from the vertical inner and outer reactor blanket to a steam generator in a closed system operating at 13.8 MPa (2000 psi) utilizing much the same conditions and components found in LWR-PWR type technology. The superheat system utilizes the steam generated on the secondary side of the pressurized water loop to feed into the reactor upper and lower superheat blanket which then feeds directly into the turbine at 8.6 MPa (1250 psig) 420°C. This circuit is comparable to the boiling water reactors in the sense that modest amounts of radioactivity will be introduced into the turbine system.

A flow diagram is shown in Figure II-4 depicting several of the major features and conditions. The system as shown is cursory pending optimization of the thermal-hydraulics and satisfactory stress analysis. Anticipated favorable conversion efficiencies with the stainless steel blanket limited to 550°C peak temperature may go as high as 40%. Blanket material optimization with steam as a coolant may substantiate even higher efficiencies for the EPR and demo plants.

D. EPR Tritium Facility Scoping Studies

R. G. Clemmer and V. A. Maroni, EPR Project

The technical basis for the design of the ANL/EPR tritium handling systems has been described in previous publications.[1,4,5] In essence, these systems must provide for (1) recycle and processing of the plasma exhaust, (2) recovery and consolidation of tritiated wastes, and (3) emergency air handling and detritiation of the reactor building and other major facility enclosures. An important attribute of the design is provision for multiple levels of containment around all components and hardware that contact sizeable quantities of tritium.

A layout of the tritium handling and fuel processing facility for a near-term tokamak reactor like the EPR is illustrated schematically in Figure II-5, and key parameters are summarized in Table II-2. Unburned DT fuel plus impurities collected by the torus evacuation system are transferred to a consolidation manifold. The gas mixture is then compressed to 0.1 MPa by a series of oilless metal bellows pumps. The fuel is purified by trapping the condensible impurities at 30 K. The trapped impurities are processed to recover residual tritium by hot gettering or by a series of cryochemical separations.[6] Helium is removed by a stripping column. The purified fuel is

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[4] W. M. Stacey, Jr., et al., "EPR-77: Revised Design for the Tokamak Experimental Power Reactor," Argonne National Laboratory, ANL/FPP/TM-77 (1977).

[5] B. Misra and V. A. Maroni, "Isotopic Enrichment of Plasma Exhausts from Controlled Thermonuclear Reactors by Cryogenic Distillation," Nucl. Tech., 40 (1977).

[6] J. L. Anderson and R. H. Sherman, "Tritium Systems Test Assembly: Design for Major Tritium Device Fabrication Review," Los Alamos Scientific Laboratory Report LA-6855-P (1977).

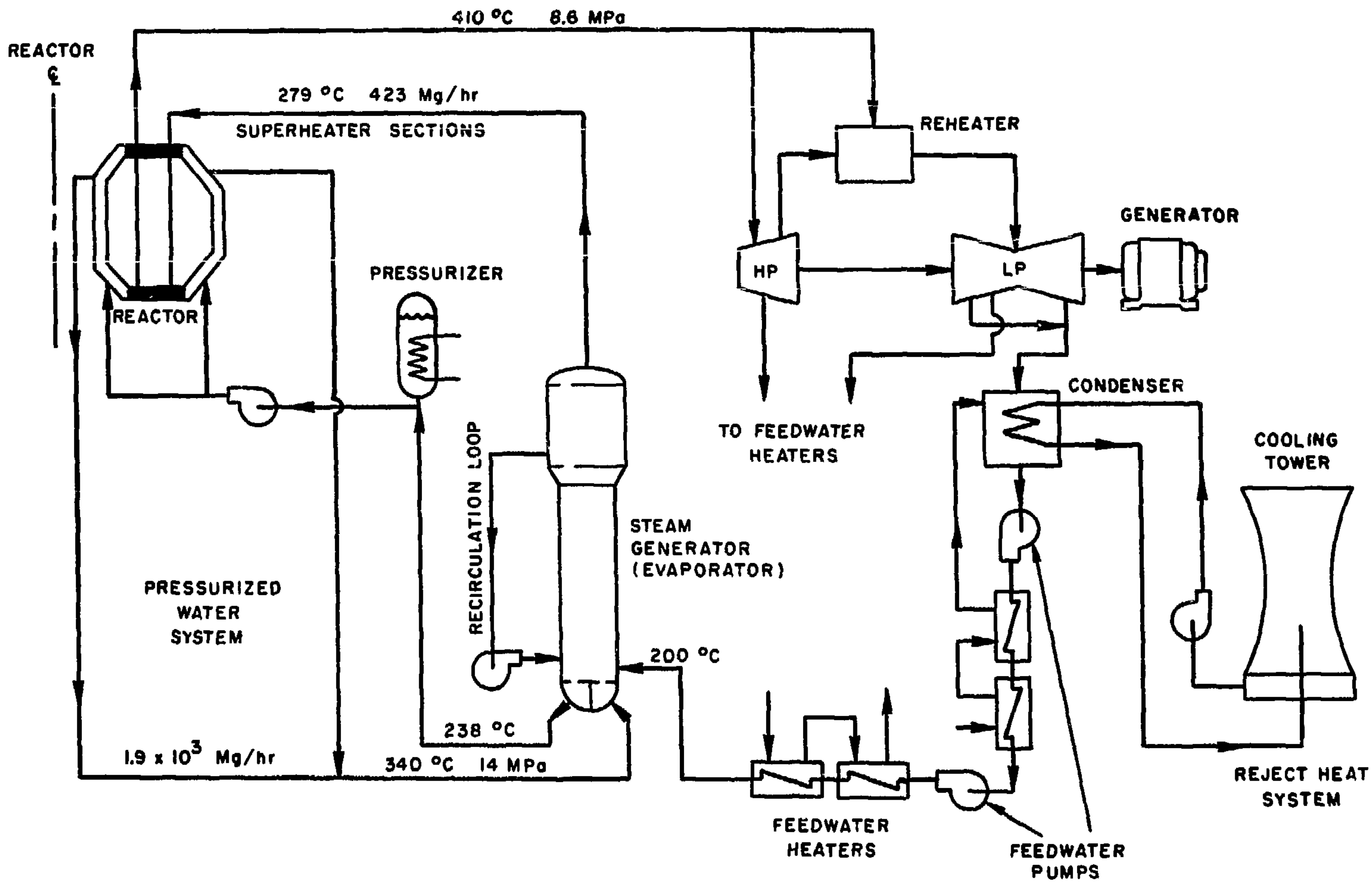
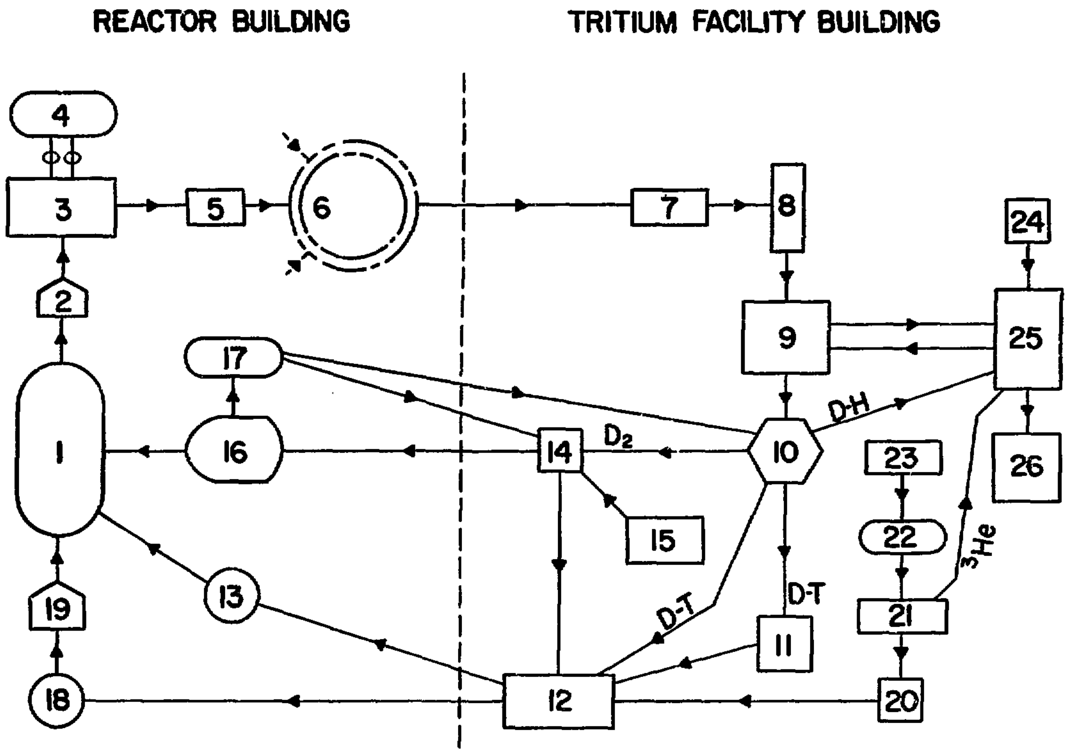


Figure II-4. Schematic of the EPR Coolant System.



1. TORUS
2. DEBRIS SEPARATOR
3. VACUUM PUMPS
4. SAFETY SURGE TANK
5. REGENERATION PUMPS
6. CONSOLIDATION MANIFOLD
7. TRANSFER PUMPS
8. COMPRESSOR
9. IMPURITY REMOVAL
10. ISOTOPIC ENRICHMENT
11. D-T STORAGE
12. FUEL BLENDER
13. COLD FUEL INJECTOR
14. D_2 STORAGE
15. D_2 SUPPLY
16. NEUTRAL BEAM INJECTOR
17. BEAM PUMPING SYSTEM
18. FUEL LIQUIFIER
19. PELLET INJECTOR
20. T_2 RECEIVER/STORAGE
21. HELIUM REMOVAL
22. HOLDING TANK
23. T_2 SUPPLY HOOKUP
24. PURGE/EFFLUENT PROCESSING
25. WASTE CONSOLIDATION
26. WASTE DISPOSAL

Figure II-5. Fuel Recycle Scenario for an EPR-Type Reactor

Table II-2. Tritium Facility and Vacuum System Parameters for the FY-1977 ANL/EPR

Vacuum System Parameters

Burn Time (s)	64.0
Dwell Time (s)	16.0
Power per Burn Pulse (MW)	325.0
Ion Density (particle/m^3)	1.30×10^{20}
Particle Confinement Time (s)	5.00
Reflectance Coefficient	0.95
Fractional Burnup	0.161
Torus Volume (m^3)	450.0
Torus Surface Area (m^2)	430.0
Number of Ducts and Pumps	12
Duct Diameter (m)	0.80
Duct Length (m)	6.00
Speed/Pump (m^3/s)	80.0
Pump Capacity (Pa-m^3)	8850.0
Preburn Pressure (Pa)	4.00×10^{-3}
Post Burn Gas Load (Pa-m^3)	4.90×10^{2}
Required Pump Speed (m^3/s)	37.05
Required Pump Capacity (Pa-m^3)	7350.0

Mass Flow Rates (g/day)

	D	T	He	H
Cold Gas Load	58.10	126.36		
Fuel (Pellets)	55.46	120.62		
Inject (Neutral Beams)	51.10			
Burn	26.50	39.75		
Exhaust	138.15	207.23	53.00	1.65
Neutral Beam Pumps	99.47	2.60		< 0.50

Tritium Inventories (g)

Torus Evacuation Pumps	20.1
Storage (30 days burn)	1192.5
Surge Tank	3.5
Liquefaction	7.1
Distillation Columns Holdup	10.2
Fuel Preparation	110.3
Total	~1334

isotopically enriched by a cryogenic distillation cascade,[5] then prepared for delivery to the torus as ambient gas, frozen pellets, or neutral beams.

Every effort is made to minimize the likelihood and consequences of leaks. The tritium handling components and hardware are enclosed in gloveboxes or are jacketed in pipe casings. Further, the tritium processing components indicated in Figure II-5, and the emergency air detritiation systems are located in a separate reinforced containment building. The jacketed pipes and the gloveboxes are purged with an inert gas stream which is processed by a tritium waste treatment system of the type described in Reference 6.

The total tritium inventory is ∿ 1300 g. Of this amount, only the 130 g that is present in the vacuum pumps and the fuel preparation units during normal operation could conceivably be released to the facility buildings (volume = 6×10^4 m^3). If this release should accidentally occur, the 50 m^3/s emergency cleanup system would take about 50 hours to reduce the ambient tritium levels to the recommended radiation control guideline of 5 $\mu Ci/m^3$.

The estimated cost of the tritium handling facilities (based in part on material in Reference 6), excluding the torus evacuation systems, in 1977 dollars is \$18 $\times 10^6$. This figure includes an overall contingency of 21%. The costs are dominated by a few select items: emergency cleanup system = \$6.3 $\times 10^6$ + 20% contingency, waste treatment system = \$3.2 $\times 10^6$ + 20% contingency, gloveboxes and purifiers = \$2.7 $\times 10^6$ + 25% contingency. The total cost is relatively insensitive to specific reactor design parameters.

E. Vacuum Systems

J. Moenich, EPR Project

The following table reflects the watts of refrigeration required per hour to operate 80 m^2 of cryopanel in each injector of EPR-77. The cryopanel uses 20°K shielding between the 77°K and 4.2°K surfaces. Cooldown to operating temperature and regeneration requirements are not included in this table, but these needs will be supplied by the same refrigerant system as used during normal operation.

Table II-3. Cryopanel Refrigerant Requirements/Hour

	Watts at		Liters
	4.2°K	20°K	L/N_2
Radiation Losses	0.1	32.0	184.0
Thermal Loads (neutron flux)*	224.0	786.0	17.6
Conduction Losses	14.0	83.0	0.4
Heat Input (from gas load)	6.0	22.0	
TOTALS/Injector	244.1	923.0	202.0

* Based on 0.001 watts/cm^3 of panel.

Twelve injectors will be necessary for plasma heating in EPR-77. The necessary refrigeration would be delivered by four units, each capable of delivering 1000 watts at 4.2°K and 4000 watts at 20°K, and four liquifiers, each capable of delivering 850 liters of liquid nitrogen per hour. Three refrigeration units and three liquifiers will be capable of delivering the required refrigeration with some reserve. The fourth unit will be on standby and for maintenance rotation. Estimated costs for each of the four refrigerators is 4 to 6 million dollars. The estimated cost of each liquifier is $375,000.

Zirconium-aluminum getter panels are another means for pumping the neutral beam injectors. These will be especially attractive should the neutron flux into the injectors result in prohibitively high thermal loading on cryopumping panels. Sixty (60) m^2 of zirconium-aluminum panel would be required to handle the equivalent gas load. The 60 m^2 of panel would operate at 400°C and require $\approx$ 260 kW of power each hour to maintain this temperature. At the present cost of 0.0191/kW hour for operating the ZGS here at Argonne operating cost of the zirconium-aluminum panel would be $\approx$ \$5.00/hour. Once every 2 to 6 days, depending on the hydrogen equilibrium pressure, the zirconium-aluminum panel temperature must be raised to $\approx$ 700°C to reactivate the panels. Assuming one hour reactivation time and a power requirement of 1440 kW, the cost of reactivating the panel each time would be $\approx$ \$28.00. These costs are based on the use of external power for heating. Thermal loading of the zirconium-aluminum panel by the neutron flux should appreciably reduce these costs. It may be necessary to use cryopumps with the zirconium-aluminum panels to eliminate gases that are either not pumped at all or are poorly pumped by the panels. These cryopumps however present, at most, 2-1/2% of the required pumping. The use of a combination of zirconium-aluminum pumps and cryopumps would result in a substantial savings in costs over the use of cryopumping alone.