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DESIGN STUDY REPORT FOR TARGET

A 1000-Mw(e) High-temperature Gas-cooled Reactor

November 1964
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General Atomic Division
General Dynamics Corporation
San Diego, California

metadc304064

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REACTOR TECHNOLOGY
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DESIGN STUDY REPORT FOR TARGET
A 1000-Mw(e) High-temperature Gas-cooled Reactor

Prepared under
Contract AT(04-3)-167
Project Agreement No. 17
for the
San Francisco Operations Office
U.S. Atomic Energy Commission

GENERAL ATOMIC
DIVISION OF
GENERAL DYNAMICS

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I. INTRODUCTION

Effective December 1, 1962, General Atomic Division of General Dynamics Corporation was awarded Project Agreement No. 17 under Contract No. AT(04-3)-167, by the U.S. Atomic Energy Commission. The scope of work for the contract stated that General Atomic was to undertake a preliminary design study and related exploratory development program for a 1000-Mw(e), advanced, gas-cooled reactor, which would have the objective of incorporating such new design features as:

1. Breeding or near breeding of fuel in a thermal-neutron-spectrum reactor using a thorium-based fuel cycle.
2. Improved fuel elements.
3. Gas-turbine-driven primary coolant circulators powered by the main coolant helium.
4. Alternative reactor component arrangement, including separate or combined pressure vessel and heat exchanger shell, and alternative materials and methods of constructing the primary coolant containing vessel(s), shielding, and containment vessel.
5. Supercritical-pressure steam cycle.

The program is divided into two major tasks: Task I, a preliminary design study, and Task II, a related exploratory development program. Task I includes a preliminary design layout and analysis of a 1000-Mw(e) power plant, preliminary capital cost estimate, fuel cycle and operating cost estimates, delineation of schedule and cost of a research and development program, and evaluation of a prototype. Task II includes exploratory development work related to advanced fuel development, turbocirculator development, alternative fuel cycle development, and alternative materials of construction.

In preparing this report of the conduct of this preliminary design study, in order to afford the reader with an over-all perspective of a typical HTGR power system while at the same time permitting maximum AEC sponsored effort to be devoted more narrowly to the conceptual development and evaluation of a new nuclear power system incorporating the five specific design features discussed above, General Atomic elected

to make use of the operating, performance, and cost data available for existing designs of certain components of reactor equipment which had been developed by General Atomic under private programs. Such data was immediately available, currently up to date and, by virtue of the fact that such components had been developed for somewhat similar reactor conditions, capable of being used with a minimum of evaluation, adaptation and contingency allowance. The study could, of course, also have been carried out under the converse assumption that only components designed under this or other AEC programs should be used. In view of the limited time available for the Task I work, however, this could have imposed undue performance limitations, or introduced large uncertainties in cost data, or both.

Thus, notwithstanding the use in this preliminary design study of data for certain of the system components (for example, the control-rod drives, the series-steam-turbine helium circulators, and the refueling mechanism) which were privately developed, in order to expedite and facilitate the remaining work, it does not follow that the TARGET system is dependent upon these items of equipment or that other equally suitable components could not be selected and/or developed for such use.

In conformance to the terms of the contract, this report presents the results of the first phase of work on Task I, the preliminary design study.

II. SUMMARY AND CONCLUSIONS

As a part of the work carried out under Task I of the TARGET program, preliminary designs and cost estimates have been prepared incorporating alternative design features for a number of plants utilizing the HTGR system and having an electrical output of approximately 1000-Mw(e). The specifications for these plants were selected in such a manner as to allow comparison of alternative designs for many of the important features of the plant. By comparing these various cases one can see the effect of the following features on the total power cost:

1. Type of primary system pressure containment--Steel vessel versus prestressed concrete vessel.
2. Number and type of reheats--Single versus double, gas versus steam.
3. Basic type of fuel element--Fission-product releasing versus fission-product retaining.
4. Type of helium circulator--Series steam turbine driven versus turbocirculator.

As a result of these studies it has been possible to arrive at the following tentative reference design, which is illustrated in Figs. 2.1 and 2.2:

Type of Primary system pressure containment. Based on the studies carried out, it appears that the use of a prestressed concrete vessel that contains not only the reactor core but the steam generators and circulators is cheaper than a steel vessel by approximately \$17/kw (0.37 mill/kw-hr). This saving results from a number of features, including the elimination of steel shells for boilers, reduction in the helium handling system size owing to lower helium inventory, reduction in the amount of concrete required specifically for shielding, reduction in the containment vessel cost, and reduction in the interest during construction owing to a shorter construction period. As a result of these studies, a design that incorporates a concrete reactor vessel with integral steam generators and circulators has been selected for the reference design.

Fig. 2.1--1000-Mw(e) reactor with steam-turbine-driven circulator

Fig. 2.2--Steam generator loop with steam-turbine-driven circulator

Type and number of reheats. Two basic types of reheats were considered: (1) gas reheat in which the steam from the main turbine is returned to the steam generator, heated by the primary helium flow, and returned to subsequent sections of the turbine; and (2) steam reheat in which live high-pressure steam from the steam generator is passed through the tube side of a steam-to-steam shell-and-tube heat exchanger. The steam to be reheated is passed through the shell side and subsequently returned to later stages of the main turbogenerator sets. The steam that is used for heating the reheat steam is then returned to the steam generator for resuperheating to throttle conditions. With either of these systems, both single and double reheats are possible. However, after a preliminary analysis of the problem, double gas reheats were excluded from further consideration because the steam pressure in the second reheat is always lower than the primary helium pressure. Therefore, if a leak should develop in the second reheater, radioactive helium could leak into the secondary cooling system. It appears that single-gas-reheat plant power costs are 0.21 mill/kw-hr (\$11/kw) less than the cheapest steam reheat (which is a double steam reheat). The reference design is therefore based on the use of a single gas reheat.

Type of fuel element. Plant designs utilizing both fission-product releasing and fission-product retaining fuel and having either BeO or graphite spines were considered. Both of these plant designs utilize purged fuel elements. In the case of the retaining fuel, the fuel is contained in the form of coated fuel particles which release a relatively small fraction of the fission products to the purge stream. In the case of the releasing fuel, the fuel is in the form of finely divided oxides, which are arranged in such a manner as to release a large fraction (approaching 100%) of the volatile fission products. While the release of the volatile fission products does make a marked improvement in conversion ratio and fuel cycle cost, a plant design capable of utilizing releasing fuel is more expensive than one capable of utilizing retaining fuel. This comes about by virtue of the more expensive fission-product removal system needed with releasing fuel and the additional shielding and containment required to cope with the higher primary system activity. Although preliminary studies indicate that the total power cost may be lower with releasing fuel than with retaining fuel, even when the penalties on capital cost are accounted for, there are considerable uncertainties in the releasing-fuel concept, principally in the ability to achieve fission-product releases approaching 100%. As a result of these uncertainties, the reference design is based on retaining fuel.

BeO spines have been chosen for the reference design because their use results in higher conversion ratios.

The plant cost with releasing fuel is about \$6/kw higher in the above comparison, primarily due to the large increase in the cost of the fission-product removal system. Developments in fission-product removal systems may considerably reduce this differential. Although retaining fuel has been chosen for the reference design, it is recommended that work should continue on the development of releasing fuel.

Type of circulator. Two types of main helium coolant circulators were considered: a series-steam-turbine-driven circulator that is powered by a turbine located in the cold reheat steam lines, and a gas-turbine-driven circulator (turbocirculator) that is powered by expanding the primary helium coolant through a gas turbine before it flows to the steam generator. Both of these types appear to be technically feasible. The turbocirculator has the potential advantage of producing higher net plant efficiency because the pumping power required to circulate the helium need not be subtracted from the net useful power output. Based on studies to date the net plant efficiency for the turbocirculator plant is approximately 46.5% versus 45.8% for the series steam turbine. The turbocirculator results in some plant cost saving due to the reduction in steam requirements in the steam cycle. There are, however, certain parts of the plant which cost more with a turbocirculator than with a series-steam-turbine-driven circulator. The circulators themselves are undoubtedly more expensive, and some of the internal ducting also costs more. Present estimates are that the net cost for a plant utilizing turbocirculators will be approximately equal to one using series steam turbines.

The turbocirculator does offer the potential, however, of having a plant with a high pumping power while retaining high net plant efficiency. This high pumping power might be used to reduce either the size of the core or the size of the steam generator, or both. It appears unlikely, however, that a large savings in power cost can be shown by the use of the turbocirculator, even with high-pumping-power plant designs. Therefore, because of the small savings presently being projected for the turbocirculator, the small probability that large savings can be shown with a revised high pumping power plant design, and the high development cost of the turbocirculator, the reference plant design is being based on the use of a series-steam-turbine-driven circulator rather than a turbocirculator.

Thus, the recommended reference design (referred to as Case 3 in the text) would incorporate the following design features: a concrete reactor vessel, a series-steam-turbine-driven circulator, a steam cycle utilizing a single gas reheat, and graphite fuel elements containing BeO central spines and fission-product retaining fuel.

For the reference design, the core consists of nearly 5500 4.5-in. diameter by 20-ft long fuel elements arranged vertically on a closely packed triangular pitch. The core diameter is 31 ft and the active height is 15.5 ft. Core inlet and outlet gas temperatures are 720°F and 1470°F, respectively. The core, which is surrounded on the sides by a 3-ft thick graphite reflector, together with the steam generator and helium circulators, are located in a prestressed concrete reactor structure having inside dimensions of approximately 40 ft by 56-1/2 ft diameter. Reactor control and refueling is achieved through 127 nozzles located in the top of the reactor vessel.

Main turbine throttle conditions are 3500 psig and 1050°F with a single 1050°F reheat. The net plant efficiency is 45.8%, corresponding to a heat rate of approximately 7500 Btu/kw-hr.

Based on the studies carried out in this program, it appears that the capital cost for a 1000-Mw(e) HTGR calculated under AEC ground rules should be approximately \$119/kw (see Table 4.18) and the fuel cost, assuming that future developments reduce the fabrication cost, as low as 1/2 mill/kw-hr (see Figs. 4.22 through 4.27). Therefore, the total power cost using 14% fixed charges on the capital should be as low as 3.2 mills/kw-hr. The reactor is capable of operating at conversion ratios as high as 0.91 using all-graphite fuel elements that retain the fission products, and 0.99 using BeO-spine fuel elements that retain the fission products, and 1.07 when operating with fission-product releasing fuel elements containing BeO. Thus, the HTGR system shows excellent promise of achieving breeding or near breeding at economically attractive power cost.

Note that the cost and conversion ratio figures given for the reference design and in Table 4.18 are for arbitrarily chosen conversion ratios with consistent fuel cycle costs. The figures given above are different in that they are intended to illustrate the highest conversion ratios and lowest fuel cycle costs which occur with different fuel element lifetimes.

III. REFERENCE DESIGN DESCRIPTION

During the past nine months, work has been directed towards establishing a reference design for a TARGET power plant. As a result of this work, all the pertinent design features for a reference design plant have been established. The design is shown in Figs. 2.1, 2.2, and 3.1 through 3.6. The reference design, which is Case 3 of the cases studied, is compared with other cases in Section IV.

The layout of the TARGET 1000-Mw(e) power plant is generally very similar to that of any other nuclear power plant. It consists of a building containing the turbogenerator set and all its auxiliaries, a building containing the nuclear steam-raising plant (i. e., the reactor, the steam generator, and their auxiliaries), and a building for reactor services.

The turbine-generator set is conventional, except that it uses supercritical steam pressure which is somewhat unusual for nuclear power plants. As is often done, in the turbine-generator set the last two stages of the high-pressure turbine are removed to allow the use of series-steam-turbine-driven primary coolant circulators. The nuclear steam-raising system, however, is somewhat different from systems used to date in nuclear power plants in that the steam generators, circulators, and core are all integrated into the same pressure vessel. This system is possible as a result of the use of a prestressed concrete reactor structure and results in considerable saving in cost and space. Six steam generator-circulator loops are utilized.

A large part of the economic saving is in the considerably simplified secondary containment for the system, which is a direct result of the use of the prestressed concrete pressure vessel and the integrated system. This integrated layout is arranged in the form of a centrally positioned core surrounded by an annulus containing the circulators and steam generators. Equally spaced around the top of this annulus are six 8-ft diameter holes through which the circulators and steam-generator components can be inserted and removed. The circulators are mounted at the lower ends of these openings with steam-generator tube bundles on either side.

The hot gas leaving the top of the core (see Fig. 3.4) passes directly down to the steam generator's superheater or reheater tube-bundle inlets. After passing through the tube bundles, the gas flows upwards, then down through the compressor and central diffuser to return to the bottom of the core.

Reactivity control is achieved by means of 242 control rods that are moved vertically upward out of the core. The rods move within control-rod guide tubes that occupy fuel-element positions within the core. The rods are moved by 121 winch-operated drive mechanisms (two rods per mechanism) that are located in nozzles above the core.

A control-rod drive has been designed as part of a program on other reactors. A prototype of this design has been manufactured and is now being tested as part of the same program. Preliminary tests are very satisfactory, and the design would appear ideally suited to the TARGET reactor.

Fuel handling is achieved through the same nozzles that house the control-rod-drive mechanisms. Fuel is handled only after the reactor has been shut down, cooled, and the helium pressure reduced to slightly less than one atmosphere. Approximately 50 fuel elements are removed from each nozzle, and approximately 18 nozzles will be used during each refueling.

As with the control-rod drives, fuel handling equipment design and development is being carried out under a program for other reactors. On the fuel transfer machine, design is complete and the manufacture of a prototype is well under way. It is scheduled for completion towards the end of fiscal year 1964, with development and testing complete around the middle of fiscal year 1965.

Conceptual designs for the remainder of the fuel handling equipment, including the fuel transfer cask, shipping cask, and storage vault, have been completed under another reactor program. Apart from the storage vault, which will require enlarging for TARGET, the equipment should be adaptable for use in either TARGET or its prototype. Some modification and development would, of course, be required.

Power Plant characteristics are given in Table 3. 1.

Fig. 3.1--Plot plan of the
1000-Mw(e) reactor

Fig. 3.2--Plant arrangement of the 1000-Mw(e) reactor

Fig. 3.3--Plant arrangement at operating floor of the 1000-Mw(e) reactor

Fig. 3.4--1000-Mw(e) reactor assembly layout (section A-A)

Fig. 3.5--1000-Mw(e) reactor
assembly layout (plan view)

Fig. 3.6--1000-Mw(e) reactor assembly layout (section B-B)

Table 3.1

POWER PLANT CHARACTERISTICS

Core

Coolant	Helium
Fuel	Coated uranium and thorium carbide particles
Moderator	Graphite and beryllium oxide
Reflector	Graphite
Average power density	7 kw/l
Heat output	2270 Mw
Void fraction	16%
Active core height	15.5 ft
Number fuel elements	5489
Core diameter	31.1 ft
Fuel element diameter	4.53 in.
Fuel element length	20 ft
Fuel element pitch	4.7 in.
Helium flow	8.32×10^6 lb/hr
Core inlet pressure	450 psia
Core inlet temperature	720°F
Core outlet temperature	1470°F
Core pressure drop	3.9 psi
Number of control rods	242
Fuel loading:	
U ²³⁵	2374 kg
Thorium.	86,330 kg
Peak-to-avg. power ratio	1.3.
Maximum heat flux	140,000 Btu/hr-ft ²
Nominal max. fuel temp.	2732°F
Conversion ratio	0.956
FIFA	1.85

Reactor Vessel

Material	Prestressed concrete
Inside diameter	56 ft 6 in.

Table 3.1--continued

Outside hexagon.	86 ft across flats
Inside height	40 ft
Outside height	76 ft 8 in.
Weight	27,000 tons
Steel liner thickness	3/4 in.
Prestressing tendon diameter	1-5/8 in.
Total length of tendons	760,000 ft

Leak Collector

Type	Leak collector
Diameter	106 ft
Height.	86 ft
Thickness of leak collector containment	0.375 in.

Steam Generators

Number of loops	6
Number of evaporator economizers . . .	12
Number of superheaters	6

Quantities for One Bundle

	<u>Evaporator- Economizers</u>	<u>Super- heaters</u>	<u>Reheaters</u>
Number of tube passes . .	8	2	2
Number of tubes	375	564	430
Tube outside diam., in. . .	0.75	0.625	1
Pitch, in.	0.937	0.843	1.343
Material.	ferritic steel	austenitic steel	austenitic steel
Tube bundle width, in. . .	48	58	58
Tube bundle depth, in. . .	51	20	26
Tube bundle length, inc. headers, ft	33	33	33
Helium pressure drop, psi.	1.8	0.5	0.5
Weight, lb	73,400	26,800	28,300
Effective transfer area, ft ²	12,660	3,970	4,840

Table 3.1--continued

Circulators

Type	Axial-flow compressor
Drive	Steam turbine in reheat inlet line
Number of circulators	6
Number of compressor stages	2
Number of turbine stages	1
Compressor pressure rise	8 psi

Main Turbines

Throttle steam pressure	3500 psig
Throttle steam temperature	1050°F
Reheat steam temperature	1050°F
Throttle steam flow	6.89×10^6 lb/hr
Condenser pressure	1.0 in Hg abs
Turbine arrangement	CCQF 52 in.
Net station output	1040 Mw(e)
Net station efficiency	45.8%

3. 1. CORE

The active core of the TARGET reactor is approximately cylindrical in shape with an effective diameter of 31 ft and a height of 15.5 ft. Surrounding the active core are graphite reflectors approximately 2.5-ft thick on the top, 2-ft thick on the bottom, and 3-ft thick on the sides. The basic core component is the fuel element, which has the outward appearance of a graphite cylinder 4.5 in. in diameter by 20 ft long. Guide tubes for control rods are also located within the active core. These are made of graphite and are similar in outside shape to the fuel elements.

The reference design core consists of 5,489 fuel elements and 242 control rods, all oriented vertically in a triangular array. The pitch of the triangular lattice is 4.70 in. Helium coolant flows upward in tricuspid channels formed by the fuel elements. The core is divided into 127 zones of nominally equal size by imaginary boundaries (see Fig. 3.7). Access to these 121 zones for refueling purposes is through the corresponding control-rod-drive nozzles in the pressure vessel. Access to the other 6 zones is through 6 auxiliary refueling nozzles. Zones away from the core boundary contain 47 fuel elements in addition to two control rods. Zones adjacent to the reflector are of somewhat different size, six of them being effectively half-zones that contain no control rods. Spacer rings on the elements are located at three axial locations to keep the elements uniformly separated. The top and bottom reflectors are integral parts of the fuel element. All of the core components are supported by a steel core support plate. Standoff pins, which are screwed into the core support plate, serve to locate and support the fuel elements as well as to conduct purge gas from the core.

The tentative reference design fuel element is a purged type that utilizes fission-product-retaining particles. The particles are packed in columns within the graphite body of the element. Beryllium oxide is utilized as an element spine material and provides a substantial gain in the moderating characteristics of the core. The complete fuel element is shown in Fig. 3.8. The main components of the fuel element are an upper reflector piece, a multihole graphite body forming the main structure, a bottom connector fitting, a beryllium oxide spine, and an internal fission-product trap assembly.

The fuel element spacing within the core is maintained by spacer rings which are raised surfaces located on the cylindrical section of the element. In the reference design, spacer rings are located at three axial positions. One is located 1 ft 8 in. from the top of the element, and the other two are at positions approximately $1/3$ and $2/3$ of the way up from

the bottom of the element. As shown in Fig. 3.8, the top spacer ring is 4.70 in. O.D., and the lower two are 4.69 in. O.D. The top spacer ring is machined onto the upper reflector piece, and the lower spacer rings are molded onto or attached to the extruded graphite body of the element.

The fuel particles are pyrocarbon coated carbide particles. Two particle diameter size ranges are envisaged. One size range is from 350 to 500 microns and contains only thorium, or thorium plus recycled uranium. The other size range is from 150 to 250 microns and contains makeup U^{235} particles, either with recycled uranium or by itself. The coatings on all the particles are expected to be approximately 100 microns thick. The two particle size ranges would be separated by screening during fuel reprocessing. As an alternative, the separation of recycled fuel from makeup fuel can be accomplished by loading the makeup fuel in a fraction of the holes in the graphite body. In this latter case, all of the particles can be of a uniform size range of 300 to 500 microns diameter. The void fraction in the packed bed is considered to be 30 vol-%.

The first point of holdup for fission products is within the coated fuel particles. Fission products that escape from the coated fuel particles are swept into an annulus between the graphite body and the beryllium oxide spine. The purge gas enters the fuel element through pores in the graphite body. The direction of flow of the purge gas is radial as it enters the element. The greater portion of the purge gas flows downward in the annular gap between the graphite body and the beryllium oxide spine. A fraction of the purge gas flows downward within the packed bed of coated fuel particles. The complete purge flow passes through the internal fission-product trap as the gas streams pass downward towards the standoff pin. Fission products that are held up in the internal trap include iodine, bromine, tellurium, cesium, strontium, and barium. The standoff pin is a hollow stainless steel cylinder that carries the fission products from the fuel element to a manifold system within the core support plate.

The graphite body is a multihole precision extrusion 15-3/4 ft long. The only machining performed on this component is expected to be on the threaded joints at either end and the possible cleaning up of spacer-ring surfaces. Spacer rings located approximately in the midspan of the graphite body and near the lower end are expected to be applied by one of two possible methods, both of which are under development. One method is to wrap the unbaked green material immediately following extrusion with a laminated graphite-cloth and pitch mixture. During baking and final graphitization of the body, this mixture will also become graphitic and result in a raised portion on the graphite body. Some

Fig. 3.7--Core and reflector layout

Fig. 3. 8--BeO spine fuel element

final machining may be necessary to smooth off the laminated structure. The second method for applying spacer rings is to allow for expansion room in the mold in which the extrusion is baked. The material will expand to fill this mold, thus leaving a section having a raised diameter.

The finished graphite body, which is 15-3/4 ft long, is joined to the graphite upper reflector by screw threads and is joined to a threaded coupling at the lower end. The coupling, a relatively short piece machined from graphite, is threaded to the bottom connector fitting. The bottom connector fitting houses the internal-trap assembly. The coupling between the graphite body and the bottom connector is required only to allow for assembly of the internal trap. Graphite screw threads are used in the Peach Bottom fuel elements and considerable joint development data have been obtained through that program. Results indicate that this type of joint is both a satisfactory and practical means of joining graphite fuel element sections.

The use of an extruded graphite body as the main structural member of the fuel element presents a number of possible advantages. First, the initial fabrication cost is expected to be low compared to other methods because of the reduced machining requirements and, also, because considerably less graphite base stock material is consumed in forming the extruded body compared with machining a body from solid graphite stock. With the packed-bed concept in conjunction with an extruded body, the requirements for holding dimensional tolerance are considerably lessened because the packed bed of particles can assume any shape in the element. The surfaces can be relatively rough and the tolerance relaxed. The use of radial purge flow allows for a relaxation of the permeability requirement for the graphite body. The helium permeability value required for this concept is approximately $1 \times 10^{-1} \text{ cm}^2/\text{sec}$ measured with helium at room temperature and 1/2 atm pressure. However, the use of a relatively porous graphite requires that pore structural characteristics be controlled rather closely so that the material is effectively backswept by inleaking helium. This insweep lowers the effective escape fraction from the body of the fission products released from fuel particles. The multi-hole body is a relatively sound member structurally with a good resistance to bending loads. Some bending is expected to be induced in it by the actions of thermal gradients and fast-flux gradients.

The beryllium oxide spines are pressed and sintered bodies that are stacked up within the graphite body. These spines fit very loosely within the graphite body for two reasons. The first is that a purge-gas flow passage must be provided by the annulus between the spine and the graphite body. The second is that a relatively large gap is required to allow for thermal expansion differences between the graphite and the beryllium oxide.

The shape of the outside of the fuel element is dictated to some degree by fuel-handling requirements. A grappling knob at the upper end of the element is provided for engagement with a fuel-handling tool for direct vertical removal of those fuel elements that are immediately adjacent to the control rods. In addition, the reduced-diameter section that is located approximately 1 ft above the bottom end of the fuel element provides for engagement with a lifting tool to raise and remove the remaining fuel elements located one or more lattice spaces away from the control rods.

In the design of the TARGET fuel elements, considerable attention has been given to the problem of fuel reprocessing. In this reference design the use of columns of loose particles appears especially promising for reprocessing. For example, if the element were sawed apart near the bottom end, the fuel particles could be ejected from the graphite body. In addition, the beryllium spines could easily be pushed out of the graphite body for reprocessing. One possible nuclear requirement of the TARGET reactor is that converted U^{233} be separated from U^{235} within the fuel elements in order to allow for separation during fuel reprocessing. Two ranges of particle size might be required with the segregated fuel. One size range of particles could be all thorium, for example, and the other size range of particles could be recycled U^{233} and U^{235} . In both cases the particles would be coated with pyrocarbon or a pyrocarbon-carbide lamination. During fuel reprocessing, the size ranges of particles could be separated by screening and the different sized particles processed separately.

The fuel elements are restricted from moving within the core by the action of their weight, by the restraint of the standoff connectors, and by holding action of a metal core-restraining barrel. Each fuel element weighs approximately 250 lb, and this weight is carried by a standoff support pin for each element. Lateral movement of the fuel elements is restricted at the lower end of the core by the standoff pin connectors. The portion of the fuel element that slips over the standoff pin is approximately 10 in. long. The strength of this connection is sufficient for each fuel element to stand vertically by itself within the core.

At the operating temperature of the core, the clearance between the fuel element and the standoff pin is approximately 0.001 in. This close fit allows negligible lateral motion at the lower end. At the upper end of the core each fuel element is restrained by lateral contact forces from adjacent elements. These forces are developed by the natural bow which will exist in each new fuel element and by the bow induced by the thermal flux gradients within the core. The forces are transmitted from one

element to another by the spacer rings. At the outer boundary of the core the forces are exerted against the side reflectors, which are in turn restrained by the metal core-support barrel. There is no upper grid plate or other mechanical supporting structure within the reactor core.

The fuel elements are designed to generate approximately 2270 Mw of thermal energy. The core power density is 7 kw per liter. The heat generated in the fuel-particle bed flows radially through the bed to the graphite body and thence to the main coolant. A radial temperature profile across the element is shown in Fig. 3.9. The thermal conductivity of the packed-bed particles is an important parameter in maintaining maximum fuel temperatures within desirable limits. Limits must be imposed on the maximum fuel temperature because of two effects. The highest limit is imposed by uranium-thorium migration through the fuel-particle coatings. This limit, which is both time and temperature dependent, is not known precisely, but it is probably greater than 1600°C for the type of particles and the time contemplated here. A second limit is imposed by the gaseous-fission-product release characteristics of the particles. The condition posed as the limit in this case is that not more than approximately 1% of all the fuel particles operate at a temperature above 1500°C at any given time, with a local maximum value of 1550°C for periods of not more than two years.

Although the present reference design is based on the use of coated particles that are relatively retentive of fission products, the ultimate TARGET core may utilize fission-product releasing fuel (i. e., fuel from which a large fraction of the volatile fission products are intentionally released). The use of such fuel, provided fission-product releases approaching 100% with sufficiently rapid release times can be achieved, results in a substantial reduction in fuel cost and an increase in conversion ratio. The advantages provided by a releasing fuel are at least partially offset by a significant increase in fission product handling problems and an attendant increase in cost for the helium purification system. However, more research and development is necessary, primarily on the ability of a core to release large fractions of the volatile fission products, before the reference core design can be based on the use of releasing fuel.

If releasing fuel were used in conjunction with BeO , the fuel element would probably look much like the existing reference design, except that the bed of loose coated particles would be replaced by a graphite matrix containing uniformly dispersed, small (diameter of 1 micron or less) particles of uranium and thorium dioxide.

Fig. 3. 9-- Radial temperature distribution for 4.5 and 4.6 in. dia. BeO-spine packed-bed fuel element containing coated (U, Th)C₂ fuel particles

3.2. CORE NUCLEAR DESIGN

Essentially all nuclear design work on the TARGET plant, as well as on other HTGR plants, has been based on the use of a uranium-thorium fuel cycle. This cycle contemplates utilizing highly enriched U^{235} and thorium in the initial core of a reactor. The U^{233} formed as a result of neutron capture in thorium is partly burned in situ and is better fuel than U^{235} by virtue of its higher η . The U^{233} (which is mixed with U^{235} and other uranium isotopes) remaining in a spent core can either be refabricated into a new HTGR core, or it can be sold and the HTGR refueled with fresh U^{235} .

Were it not for the buildup of neutron poisons such as U^{236} and Np^{237} , which result from neutron capture in U^{235} , the use of recycled fuel would result in a considerable improvement in neutron economy compared to U^{235} . As a result of the buildup of these isotopes, a continuous recycle scheme in which all of the spent fuel is reprocessed, mixed with the required amount of U^{235} makeup, and refabricated into new fuel elements, does not appear optimum. However, several other recycle schemes that retain most of the advantages of the higher η of U^{233} and that limit the buildup of U^{236} and Np^{237} have been conceived. All of these schemes involve segregation of the thorium, the recycled uranium, and the U^{235} makeup. If the segregation is limited to two parts, then there are three types of recycle:

<u>Type</u>	<u>A Particle</u>	<u>B Particle</u>
I	Th	Recycled U + U^{235} makeup
II	Th + recycled U	U^{235} makeup
III	Th + U^{235} makeup	Recycled U

In all cases, the spent B particles are sold where the uranium formed in the A particles is recycled either in A particles or B particles, depending on the type of recycle. The advantages and disadvantages of these three types of recycle are discussed in Section 4.3.2.

The results of fuel cycle calculations on the reference design core (BeO spine with fission-product retaining fuel) are presented in Table 3.2 and 3.3 for Type 1 and Type 2 recycles respectively. These calculations are based on a Be/Th ratio of 28, which results in a BeO loading of 250.3 tonnes and a C loading of 308.2 tonnes. Both these tables assume the standard GA cross sections (see Section 4.3.2.). It should be noted that in the simplified cost calculations used, no deductions are made for U^{232} content of the fuel and no credit is taken

Table 3. 2

RESULTS OF FUEL-CYCLE STUDIES FOR TYPE 1 RECYCLE

(Packed-bed Fuel, Be/Th = 28, GA Cross Sections)

	4-year Life	5-year Life	6-year Life	7-year Life	8-year Life
Initial Be/U ratio (fissile U only)	962	871	785	705	635
U ²³⁵ feed/cycle, kg	128	141	167	199	241
Net fissile U consumed per cycle, kg	43	68	100	136	180
Fissions per initial fissile atom	1. 287	1. 476	1. 606	1. 703	1. 759
Eta	2. 139	2. 135	2. 128	2. 120	2. 112
Conversion ratio	0. 927	0. 908	0. 885	0. 865	0. 843
Fuel-cycle cost, mills/kw-hr(e)*:					
Private fuel	1. 509	1. 378	1. 314	1. 286	1. 287
AEC fuel	1. 157	1. 011	0. 932	0. 881	0. 860

* Fabrication cost of full core = \$22, 000, 000 (= 1. 22 x \$18, 000, 000)

Table 3. 3

RESULTS OF FUEL-CYCLE STUDIES FOR TYPE 2 RECYCLE

(Packed-bed Fuel, Be/Th = 28, GA Cross Sections)

	4-year Life	5-year Life	6-year Life	7-year Life	8-year Life
Initial Be/U ratio (fissile U only)	886	811	743	676	614
U ²³⁵ feed per cycle, kg	77	114	154	202	254
Net fissile U consumed per cycle, kg	55	87	125	170	220
Fissions per initial fissile atom	1. 075	1. 231	1. 352	1. 435	1. 490
Eta	2. 131	2. 128	2. 107	2. 098	2. 088
Conversion ratio	0. 909	0. 883	0. 859	0. 835	0. 815
Fuel-cycle cost, mills/kw-hr(e)*:					
Private fuel	1. 670	1. 539	1. 471	1. 449	1. 451
AEC fuel	1. 247	1. 099	1. 015	0. 967	0. 940

* Fabrication cost of full core = \$22, 000, 000

for production of Np^{237} which is the source material for Pu^{238} , which in turn is desirable for space application; fabrication costs are independent of loading and of the method of recycle employed. The cost figures are therefore to be taken only as rough guides; in particular, the comparison of the two recycle methods is not adequately represented.

The results of Tables 3.2 and 3.3 indicate high conversion ratios and lower fuel cost for Type 1 recycle than for Type 2 recycle. By using Type 1 recycle, a conversion ratio of 0.843 can be achieved at a fuel cost of 0.86 mill/kw-hr using an eight year core life, or a conversion ratio of 0.927 at a fuel cost of 1.157 mills/kw-hr with a four year core life. The effect of other core fabrication cost and other assumptions on cross sections is shown in Section 4.3.2.

The above results are based on the use of a fuel in which most of the fission products are retained. If the volatile fission products can be substantially all released from the core, a decrease in fuel cost and an increase in conversion ratio to greater than 1.0 is possible (see Section 4.3.2).

3.2.1. Fuel Cycle Costs (Assumptions)

The fuel cycle costs were estimated in the standard manner* and the individual items are enumerated below:

1. Fabrication. The cost of fuel fabrication and delivery to the reactor site was assumed to be independent of fuel loading within the range considered. Three values were assumed for the all-graphite core: \$25 million, which roughly corresponds to present costs; \$18 million, which would apply in the near-term future; and \$10 million, which is estimated as a future cost. For fuel elements with BeO spines, these costs were increased by 22%. In all cases, the core costs include the use charge on the fissile material prior to delivery to the reactor site and the losses in fabrication. The near term cost is based on General Atomic's experience in other HTGR programs; the future cost is a logical extrapolation from this experience.

2. Irradiation Inventory. The fuel was assumed to be stored at the reactor site for 30 days before being inserted. As is standard practice under Government ownership of fuel, a 4-3/4% use charge was paid each six months on the fuel which was present in the reactor. Under private ownership, a 10% use charge was assessed against all fuel obtaining in the reactor.

3. Postirradiation Inventory. The postirradiation inventory was assessed against the fuel discharged from the reactor during the four-month cooling period and the six months occupied in shipping and processing.

*Kaiser Engineers, Fuel Cycle Costs, TID 7025, Vol. 4, March 15, 1962.

4. Burnup Charges. The standard enriched uranium price schedule* was used for U^{235} (\$12/g of U^{235} when enriched to greater than 90%) and \$14/g of U^{233} was assumed when enriched to greater than 90%. Pu credit is negligible.

5. Shipping. A constant cost of \$120,000/core was used for shipping irradiated fuel to the chemical processing plant.

6. Chemical Processing. The one-ton per day standard chemical plant was assumed; however, the throughput of fully enriched uranium was reduced to 30 kg/day rather than the usual 44 kg/day. This is based on an estimate made by ORNL†. The chemical plant cost was taken as \$17,000/day, with a minimum turn-around time of 2 days and a maximum turn-around time of 8 days.

3.2.2. Fuel Cycle Costs (Results)

The variations in fuel cycle cost and conversion ratio are shown in Figs. 3.10 and 3.11 as a function of fuel loading and cycle time. For a \$22 million core, using BeO spines, Type 1 recycle, fission-product retention, and Government ownership, it is clear that minimum costs require long cycle time and a Be/Th = 32 for GA cross sections and a Be/Th = 28 for modified GA cross sections (see in Section 4.3, "Nuclear Data"). In contrast, maximum conversion ratio requires short cycle times and lower Be/Th for both sets of cross sections. With private ownership of fuel, these differences are magnified, principally with respect to Be-Th ratio. The detailed cost breakdown for the Be/Th corresponding to the various cycle times for minimum fuel cycle costs are shown in Table 3.4 and for maximum conversion ratio are shown in Table 3.5. In all cases, GA cross sections were used and fission products were assumed to be retained. That the fuel cycle costs monotonically decrease with increasing cycle time is clearly due to the steady decrease in the fabrication, shipping, and reprocessing costs, which are inversely proportional to cycle time. The net burnup costs steadily increase with cycle time owing to the reduction in conversion ratio. For both sets of cross sections, it appears that cycle times in excess of 8 years might not be economically justifiable, and using a 6-year cycle time increases the costs by less than 0.1 mill/kw-hr. The detailed cost breakdown for the fuel loading corresponding to minimum fuel costs with a 6-year cycle time are shown in Table 3.6 for gaseous fission-product release, for modified GA cross sections, and for an all-graphite core. Gaseous fission-product release

* Federal Register, Enriched and Depleted Uranium Base Charges, FR Doc 62-5124, May 28, 1962.

† S. Jaye, et al., A Study of the Fuel Value of U^{233} , ORNL-CF-60-4-79, April 11, 1960.

Fig. 3. 10--Variation of fuel cycle cost and conversion ratio with fuel loading and cycle time using GA cross sections, fission-product retention, and core cost = $\$22 \times 10^6$

Fig. 3.11--Variation of fuel cycle cost and conversion ratio with fuel loading and cycle time using modified GA cross sections, fission-product retention, and core cost = $\$22 \times 10^6$

Table 3.4

BREAKDOWN FOR MINIMUM FUEL COST AT VARIOUS CYCLE TIMES*

	3-year Cycle	4-year Cycle	5-year Cycle	6-year Cycle	7-year Cycle	8-year Cycle
Be/Th	40	36	36	32	32	32
Costs, mills/kw-hr:						
Fabrication	0.97	0.73	0.58	0.48	0.42	0.36
Shipping	0.05	0.05	0.03	0.03	0.02	0.02
Chemical processing	0.09	0.07	0.06	0.06	0.05	0.04
Processing losses	0.02	0.01	0.01	0.01	0.01	0.01
Use during irradiation	0.15	0.15	0.15	0.18	0.18	0.19
Use after irradiation	0.01	0.01	0.01	0.01	0.01	0.01
Net burnup	0.09	0.11	0.14	0.14	0.18	0.21
Total	1.38	1.13	0.98	0.91	0.87	0.84

*Costs assume Government fuel ownership, GA cross sections, fission-product retention, BeO spines, core cost \$22 million.

Table 3.5

BREAKDOWN FOR MAXIMUM FUEL CONVERSION RATIO AT
VARIOUS CYCLE TIMES*

	3-year Cycle	4-year Cycle	5-year Cycle	6-year Cycle	7-year Cycle	8-year Cycle
Be/Th	32	32	28	28	28	28
Costs, mills/kw-hr:						
Fabrication	0.97	0.73	0.58	0.48	0.42	0.36
Shipping	0.05	0.04	0.03	0.03	0.02	0.02
Chemical processing	0.11	0.08	0.07	0.06	0.05	0.05
Processing losses	0.01	0.01	0.01	0.01	0.01	0.01
Use during irradiation	0.19	1.17	0.20	0.20	0.20	0.21
Use after irradiation	0.01	0.01	0.01	0.01	0.01	0.01
Net burnup	0.06	0.08	0.11	0.14	0.16	1.18
Total	1.40	1.12	1.01	0.93	0.87	0.84

*Costs assume Government fuel ownership, GA cross sections, fission-product retention, BeO spines, core cost \$22 million.

Table 3.6

BREAKDOWN FOR MINIMUM FUEL COSTS WITH A SIX-YEAR CYCLE TIME*

Special Conditions	Modified GA Cross Sections	Volatile Fission Product Release	Graphite Spines
Be/Th (or C/Th)	28	32	(170)
Costs, mills/kw-hr:			
Fabrication	0.48	0.48	0.40
Shipping	0.03	0.03	0.03
Chemical processing	0.06	0.05	0.05
Processing losses	0.01	0.01	0.01
Use during irradiation	0.18	0.12	0.18
Use after irradiation	0.01	0.01	0.01
Net burnup	0.06	0.01	0.28
Total	0.83	0.71	0.96

*Costs assume Government fuel ownership, and core cost \$22 million.

reduces the fuel cycle cost by 0.2 mill/kw-hr principally by increasing the conversion ratio and thus reducing the depletion and use charges. If the modified GA cross sections are correct, the fuel costs drop by 0.1 mill/kw-hr for similar reasons. If an all-graphite core is used, the fuel costs increase by only 0.05 mill/kw-hr because the increased depletion charges are largely offset by the reduced fabrication cost of the core.

3.2.3. Core Temperature Coefficient

The reactor temperature-coefficient calculation is both time consuming and expensive; consequently, it was only evaluated for a representative case. (The case considered had a six-year cycle time, a Be/Th = 28, standard GA cross sections, and retained all fission products.) The fuel temperature coefficient and the isothermal temperature coefficient are shown in Fig. 3.12 as a function of temperature for the equilibrium core. The fuel coefficient is primarily due to the Doppler broadening of the Th²³² resonances and therefore approaches zero when the broadened resonances overlap. Below 1200°K the isothermal temperature coefficient is dominated by Doppler effects in thorium and by the 0.08 ev resonance in Xe¹³⁵. At higher temperature, the low lying resonances in U²³³ introduce a positive contribution and result in an increasing temperature coefficient which first goes positive at 2600°K.

Fig. 3.12--Fuel temperature coefficient and isothermal temperature coefficient as a function of temperature for the equilibrium core

In the course of nearly all accidents, the fuel coefficient terminates the excursion and the magnitude of the isothermal coefficient is relatively unimportant. It is only during very slow accidents (such as loss of coolant, which is not possible in a concrete vessel) coupled with a complete failure of the control system scram that the isothermal coefficient is of importance. Even then, it would require a 0.85% Δk reactivity insertion before reaching 2600°k. One of the principal safety features of an HTGR is the difficulty of inserting reactivity. Generally, the worst accident is related to the continuous withdrawal of one control rod pair. However, in TARGET this would add only 0.2% Δk and would be fully compensated by a core temperature rise of only 150°C without any scram. Thus, the temperature coefficient prevailing in the reactor is sufficient to produce stable operation. If it were desired to make the temperature coefficient more negative, it could easily be accomplished by adding Rh¹⁰³ to the feed or by using a reduced uranium enrichment for the makeup. If 11.5 kg of Rh¹⁰³ were added to the fuel inserted each year or if the uranium makeup had an enrichment of 60% rather than 93-1/2%, the isothermal temperature coefficient would be less than -1.0×10^{-5} beyond 3000°k. Rhodium enhances the temperature coefficient at high temperatures through the 1.05 ev resonance. Reducing the enrichment is effective for the same reason but with Pu²⁴⁰ being the active nuclide.

3.2.4. Xenon Stability

A preliminary investigation of the stability of the TARGET reactor has been made using the method suggested by Randell and St. John*. The reactor is stable in both the axial and azimuthal mode but will exhibit damped oscillations. In both cases, the magnitude of the oscillation will drop by a factor of e in roughly 4 hr. The azimuthal and axial oscillations have periods of 14 and 25 hr, respectively. It should be pointed out that this analysis is relatively crude and in some cases has been shown to be optimistic†. As part of the TARGET program, work is presently being directed toward a more realistic model that will yield more definitive results.

*D. Randell and D. S. St. John, "Xenon Spatial Oscillations," Nucleonics, March, 1958.

†R. S. Wick, Space and Time Dependent Flux Oscillations, WAPD-TM-138, August, 1958.

3.3. FUEL HANDLING AND STORAGE

3.3.1. Fuel Handling Process

The refueling process is started after a reactor cooling-off, coolant cleanup, and depressurizing period lasting about 24 hr subsequent to reactor shutdown. The core inlet helium pressure and temperature during refueling are approximately 14.5 psia and 250°F, respectively. Each of the 127 nozzles on the reactor vessel is utilized for refueling operations. Based on a 6-year refueling cycle, 21 nozzles are opened for refueling in an average year. Refueling is by core zone; the standard zone contains 47 fuel elements and two control-rod guide tubes. Handling, removal, and insertion operations for an individual zone are all performed through one nozzle only. In an average refueling year, about 915 fuel elements are replaced in the core in approximately 2 weeks.

The spent fuel elements are removed to the fuel storage vault (see Figs. 3.2 and 3.3) for a nominal decay period of 3 months before the shipment of spent fuel commences. The vault has a capacity of 1224 elements. In it, the elements are cooled by the natural convection of nitrogen. To inspect the fuel elements in position and during removal and insertion, a reactor viewing device is used. This apparatus is to be inserted in one of the reactor nozzles by means of the auxiliary cask and mounted on an isolation valve. The device is also designed for various aspects of internal pressure vessel inspection.

For routine and nonroutine core operations, a series of adapters and tools are provided. This includes an adapter system for the fuel transfer machine which permits rotation of fuel and reflector elements, and failed fuel element locating apparatus. Also included are various grapples for handling broken fuel elements, element fragments, fuel standoff pins, etc., and for cutting control-rod cables should this become necessary.

The fuel transfer machine is a remotely operated, electrically controlled, hydraulically powered machine. The materials in the exposed parts have been selected on the basis of a maximum gamma exposure rate of 10^6 r/hr and 600°F environmental temperature. Basically, the transfer machine consists of an upper drive compartment, a carriage assembly, a lower assembly, and a grapple jaw assembly. The machine is inserted into the nozzle with the upper end supported by, and sealing against the isolation valve. With the fuel transfer cask in position and sealing against an adapter spool, an isolation valve inside the transfer machine is opened. The main carriage is lowered from its retracted position operating in parallel with the grapple carriage, the latter of which will stop in its lowest (or grappling) position on the lower channel assembly. When the channel and the jaw carriage are in their lowest positions, the grapple jaws are opened by a separate drive.

Closing the jaws permits the element to be lifted from its standoff pin by the activation of the carriage drive, so that the jaws travel upward. When the element is free of the standoff pin, the main carriage drive is reversed to

raise the element to its upper unloading position, where it is transferred to the fuel transfer cask grapples. An element is installed in the core by reversing these operational sequences.

The hydraulic fluid for the drive motors is supplied from a separate mobile power unit that is connected to the machine with flexible lines with quick disconnect couplings. Hydraulic motors housed in the upper compartment of the fuel transfer machine are employed. All electrical wiring is distributed internally in the machine from multipin Amphenol-type bulkhead fittings, which connect the machine to its control console via cables.

Shielding is provided internally as well as externally to the upper end of the transfer machine. Part of the external shielding is contained in the built-in shielding in the reactor isolation valve. Above that, a lead sleeve, called an adapter spool, surrounds the otherwise exposed upper end of the drive section of the fuel transfer machine.

Should a fuel element become stuck in an elevated position in the transfer machine, a cooling system that utilizes a small portion of the hydraulic drive fluid is built into the upper drive section. Furthermore, additional cooling is obtained by bleeding helium into the adapter spool, over the top surface of the drive compartment, through the compartment, and then through the internals of the machine into the reactor core. A second stream of helium sweeps the area between the azimuth guide tube and the control-rod-drive nozzle before entering the reactor vessel.

Each transfer machine is controlled and operated from its own central console. The machines are portable, so they can be located in close proximity to a nozzle that is being worked in.

Electrical interlocks are provided to assure safe and proper sequencing of operations, and to prevent damage to components. All power supplies are provided with low-voltage-sensitive relays to assure that all sources are energized prior to operation of controls.

3.3.2. Fuel Transfer Equipment

The two fuel transfer casks are used to transport spent and new fuel elements between the reactor and the storage vault, and also between the storage vault and the shipping facilities.

Each cask is designed to receive and store for an indefinite period either five fuel elements plus two control-rod-guide tubes, or six fuel elements. It is mounted with all its hoisting mechanisms, cooling systems, and controls on a selfpropelled carriage that runs on tracks between the reactor control-rod-drive nozzle area and the service building. The electrical controls are primarily automatic and semi-automatic, with provisions for manual control override by the operator, whose station is at one end of the carriage, where he will be able to operate all controls.

The carriage, which is driven by a 7-1/2 hp motor, has a traveling speed of 50 ft/min. A 1/2 hp motor drives the carriage at a creep speed of 6 in. /min for positioning.

The cask is designed to handle and store fuel and to provide adequate shielding to protect the operating personnel. The shielding consists of 12 in. of lead surrounding the fuel elements. Maximum radiation level at the surface of the cask does not exceed 10 mr/hr, and at 1 m from the surface, 4 mr/hr.

Other than the shielding, the cask consists of six major components: grappler hoisting mechanism, grappler locating mechanism, fuel storage rack, shielding door, shielding skirt, and cooling systems.

The cooling system is designed as a dual system, with each system capable of handling the entire heat load of six fuel elements after a period of 12 hr after reactor shutdown. Each system consists of a closed helium loop for cooling the stored elements. In each loop, the heat is transferred to a helium-to-air heat exchanger, through which ambient air is circulated for dissipation into the surrounding atmosphere.

Interlocks are provided throughout, so that, for example, unless the positioning drives are stationary and the storage rack and lead door positioned, the grapple hoist is inoperative. The fuel transfer casks (and the auxiliary transfer cask) obtain electrical power from enclosed conductor rails located in the building floor and within the refueling bridge.

A refueling bridge is provided in order to accommodate the fuel transfer casks and the auxiliary transfer cask in the area containing the reactor refueling nozzles. The bridge travels on a set of rails and is driven through a gear reduction unit; it travels at a maximum speed of 20 ft/min. A smaller motor connected to the drive shaft via a clutch arrangement is used for creep speeds when positioning. Transverse to its running track, the bridge carries a set of rails to accommodate two casks.

The combined movements of the bridge and the equipment on top of it permit accurate equipment positioning over any one of the 121 reactor control-rod-drive nozzles and six refueling nozzles.

A control cabinet for the operator is located on the bridge structure. Interlocks are provided throughout. A 10 ft section of the enclosed conductor rails in the building floor leading onto the bridge is electrically isolated from the remainder of the track. This means that this section will not have power unless the bridge and floor rails are aligned, thus permitting an alignment switch to be thrown to energize the isolated section. Interlock cables are connected between the bridge controls and the machine located on the bridge, in order to prevent movement of the bridge when equipment is connected to a reactor nozzle. In case there should be no power on the bridge drive (either because the machine has been deliberately stopped or because of a power failure), a brake is

automatically applied. In addition to the electrical track interlock, a mechanical lock ensures that the bridge track is aligned with the floor track. As a further safety measure, a tapered wedge-type stop is provided near the end of the floor track to prevent any accidental overrun of the apparatus when the bridge is not in position. The bridge control cabinet also provides the capability of controlling the power to any machine while on the refueling bridge.

To provide a gas-tight closure on a reactor refueling nozzle, a reactor isolation valve is used. Three valves are provided with this reactor--two for fuel transfer machines and one for the reactor viewing device. The valve is a gate type with a shielded body and a motor-driven shielded gate.

Whether closed or open, and with an adapter spool and a cask in place above it, the maximum dose rate increment at the operating personnel position due to this equipment does not exceed 2-1/2 mr/hr. This limitation prevails even when a spent fuel element is passing through the open valve.

Attached to the bottom of the valve is a spring-loaded bellows, which effects a gas-tight connection to the top flange of the refueling nozzle. The valve is supported by the building structure, not the vessel nozzle, by means of built-in screw jacks which permit the leveling and adjustment of the valve elevation.

The gate drive motor, which is mounted externally, drives the gate by a screw. The gate contains an upper and lower seal plate that is actuated pneumatically for making gas-tight closures with the top and bottom openings of the valve body. Limit switches stop the drive motor at the extremes of gate travel, and a solenoid brake locks the drive screw when the gate is in either the "full open" or "full closed" position. The latter is also an interlock to prevent actuation of the seal plates except in this position. A pressure switch is provided that acts as an interlock in the drive motor control circuit to prevent opening the gate unless the seal plate actuating cylinder has been vented.

3.3.3. Fuel Storage and Auxiliary Handling Equipment

To bridge the gap between the reactor isolation valve and any one of the transfer casks used, a lead-shielded adapter spool is employed. Being basically of pipe shape, the adapter spool provides a seal-tight joint below with a reactor isolation valve and above with either the auxiliary transfer cask or a fuel transfer cask. Ordinarily, an adapter spool and a reactor isolation valve are handled as a single unit. The shielding design criteria of this item are the same as for the isolation valve, 2-1/2 mr/hr maximum dose rate increment to the operating personnel, when a spent fuel element is passed through.

An adapter spool will, during certain phases of the refueling operation, house the upper drive end of a fuel transfer machine. As such, it provides shielding for the machine at this point. Built-in facilities for supplying the machine with hydraulic power and electrical services are provided through the wall.

A periscope is also built into the wall of the adapter spool. It can be moved out from its storage pocket and its objective head can be rotated. A built-in light provides the illumination. The viewing range is such as to permit close observation of a fuel element being passed through the adapter spool, or observation of the core grid plate approximately 50 ft below. Interlocks, in the form of limit switches, are provided to protect the viewing device from damage by ensuring that equipment cannot be moved through the spool unless the periscope is stowed in the wall pocket.

For storage of spent fuel elements prior to shipment and for storage of reprocessed elements prior to insertion in the reactor, a 1224-element capacity vault is provided in the reactor service building. The capacity is based on a six-year fuel cycle. Allowance in the capacity provides for spares and temporary storage of additional elements, which may be removed from the reactor during some phases of the refueling operations.

Fuel elements and control-rod-guide tubes are stored in the vault in a dry, uncanned condition. The fuel elements are stored for a nominal minimum decay period of three months before shipment to a reprocessing plant. During this period the afterheat generation and fission-product release decay to levels that are low enough to permit shipping without resorting to complicated cooling apparatus on the shipping casks.

The vault is cooled by natural convection of nitrogen; a cleanup system removes the released fission products from the gas. Tracks from the reactor area facilitate use of the fuel transfer casks, both when servicing the reactor and when shipping and receiving fuel.

Inside the vault, fresh and/or spent elements are stored in a rotating rack that is about 30 ft in diameter by 19 ft high. Each element is stored by top loading into an individual 6-in. diameter tube. Concrete provides the shielding, limiting any personnel exposure to less than 2-1/2 mr/hr. The rack is rotated by a removable motor drive assembly at the periphery. To rotate the rack one full revolution takes about 15 min and to index one space takes about 10 to 25 sec. The rack rides on heavy duty Meehanite rollers and circular tracks.

Elements are loaded into the vault from a fuel transfer cask through one of the two vault isolation valves provided, after the access plug has been removed from a particular opening. One access opening is provided for each circular row of tubes in the rack, making a total of 14 openings. An external indexing device connected with the rack drive assures proper orientation of each tube in relation to its access opening.

Two periscopes with built-in illumination and located 180° apart permit viewing of the elements during both insertion and removal. For this purpose, the periscopes view an area about 1 ft high between the top of the rack and the bottom of the access ports. The periscopes serve as an aid in maintaining the vault inventory and are also used to observe whether or not an element has been properly seated in its tube.

The maximum afterheat generation for the maximum number of elements in the vault is about 8.4×10^6 Btu/hr. This heat is removed by the natural convection of nitrogen in the vault passing between the walls of the element support tube and the elements. The heated gas leaves from the tops of the tubes and is directed through a cylindrical fintube gas-to-water heat exchanger at the top of the vault. Maximum entrance temperature to the heat exchanger is 500°F , and the exit temperature is 150°F . A maximum of 400 gpm of cooling water removes the heat on the secondary side of the exchanger, which is divided into multiple tube rows, each plumbed separately. In the event of a leak, any row may be shut off without jeopardizing the over-all cooling effect. During loading and unloading periods, a negative pressure of 0.5 psig is maintained inside the vault. Otherwise the pressure is maintained at atmospheric. Water cooling coils embedded near the inner surface protect the shielding concrete from exceeding 150°F as a result of gamma heating and thermal heat transfer.

Forty gpm of cooling water is required in this system. With the internals of the vault being conservatively and ruggedly designed, only the plug-mounted heat exchanger and the vault drive are designed to be removed for servicing. Should decontamination be required, a small plug in the heat exchanger plug can be removed and a spray nozzle inserted. Hot drainage is provided at the bottom of the vault. For removal of the drive, the auxiliary cask and a reactor isolation valve are used.

Two fuel storage isolation valves are used for gas-tight closures on the fuel storage access openings during loading and unloading. Each valve is very similar in construction to the reactor isolation valve. When closed, it forms a gas-tight closure on the vault while at the same time the operating personnel are not subjected to an exposure rate of over 2-1/2 mr/hr. The valve body also provides similarly adequate shielding when the fuel transfer cask is in place over the valve, and a spent fuel element retrieved from the reactor is being lowered through the open valve into the vault.

The valve gate consists of a lead plug plus a toggle actuated sealing plate, which at the end of the plug travel rises and makes a gas-tight

closure over the top opening of the valve body. The gate is driven by a screw that penetrates the valve housing through a seal gland. An externally mounted gear motor drives the screw. Limit switches are provided to limit the extreme travels of the gate.

The plug removal cask is a shielded gas-tight container used for removing and installing the fuel storage vault access plugs. The cask consists basically of a shielded housing, a plunger, and a gate. The plunger penetrates the top of the housing through a sealing gland, as does the gate actuating shaft. A plug is drawn up into the cask or lowered into its vault access tube by means of the plunger, which screws into the top of the plug. A geared, manually operated cable winch is provided to facilitate the manual raising and lowering of the plunger. The clapper type gate is provided as a collector for any radioactive dust or particles which might otherwise drop off the plug during travel to the storage area. A limit switch, which is actuated when the plunger is in its up position, acts as an interlock in the control circuit of the vault isolation valve; this protects the valve gate by preventing it from being moved unless the plunger is in the up position.

A fuel handling purge system is provided to prevent release of activity to the reactor building atmosphere during refueling. Purging operations of the equipment used for fuel handling are controlled manually. All valves are air-operated and electrically controlled. Purging operations of fuel handling equipment in the reactor area are controlled from the fuel transfer machine console, which will contain the controls, indicating lamps, pressure indicating instruments, and radiation monitoring instruments.

In the storage vault area similar controls are provided on the storage vault console.

The reactor viewing device is an articulated periscope with a built-in lighting system that is used during shutdown for direct observation of the reactor interior, or for indirect observation by adding a TV camera to the eyepiece. The device fits into any one of the reactor refueling nozzles after the control-rod drive or shielding plug has been removed. When installed, it makes a gas-tight closure and provides sufficient radiation shielding to limit the dose rate to an observer to a maximum of 2-1/2 mr/hr during normal refueling operations.

The objective head has built-in motion to permit viewing over an area ranging from vertically down to vertically up and through a horizontal angle of 180°. The viewing range covered is from one foot from the objective head to the core grid plate.

The periscope consists of a vertical section that extends through the refueling nozzle and a horizontal section that can be extended to a distance of 4 ft 9 in. from the nozzle centerline. The extension is accomplished by means of a lazy-tong arrangement, with the periscope forming one leg. Image transmission at the joints of the lazy-tong is through a system of optical lenses and reversing prisms. The objective head is capable of swinging through two angles of 180° at right angles to each other. These motions are performed by small gear motors mounted on the viewing head. The horizontal lazy-tong section of the periscope can be swung through a horizontal angle of 360° . Both this action and the extension and retraction of the lazy-tong are accomplished through a system of gears and shafts which extend upward to drive motors located at the top end of the viewing device. Nonbrowning glass is used for the optical components. A picture derotation device is provided to compensate for rotation of the transmitted image caused by motion of the articulated periscope joints.

Provision is made for attachment of a TV camera to the eyepiece for viewing on a monitor, or for attaching a camera to photograph features of interest inside the reactor. Provision is also made to insert special grappling or manipulating tools alongside the periscope. Since the drive motors, which are located near the top of the viewing device, and their associated shafting have to be removed to make room for these special tools, an in-line objective head is provided to replace the articulated head during this mode of operation.

The auxiliary transfer cask serves primarily as a means for inserting and removing equipment into or out of the reactor refueling nozzles and also for transporting this equipment to and from storage. The cask is capable of handling the fuel transfer machine, the reactor control-rod drives, the reactor viewing device, and miscellaneous smaller equipment. When in position over either adapter spool, it forms a gas-tight containment, and it provides sufficient radiation shielding from control-rod drives to limit the operating personnel to a maximum exposure rate of $2\frac{1}{2}$ mr/hr.

For mobility, the cask is mounted on a motor-driven carriage running on the rail system between the reactor and the reactor service building. A two-speed drive provides a rapid transfer speed of 50 ft/min and a creep speed for positioning of 6 in./min.

A two-speed electric hoist mounted within the top section of the cask has a 5-ton lifting capacity at speeds of 20 or 7 ft/min. The hoist is equipped with a solenoid-actuated grapple, which matches a lifting adapter on each piece of equipment to be handled by the cask. The grapple is equipped with limit switches that serve as interlocks to prevent engaging the pick-up latches unless the grapple is properly seated in the

lifting adapter and that prevent disengaging the pickup latches unless the grapple has deposited its load. Limit switches built into the hoist provide interlocks to limit the up-travel of the grapppler. Other limit switches prevent high speed up and down travel when the hoist is within approximately 1 ft of the equipment in the seated position. A load cell acts as an interlock to prevent the hoist from raising if a load considerably in excess of the equipment weight is sensed. A slack cable sensing device serves as another interlock to stop the machine from lowering if the hoist cables become slack as a result of equipment being hung up during lowering. Mechanical counter read-outs, driven by a synchro system, give continuous indication on the control panel of the grapple position.

The auxiliary cask is provided with a movable shielded skirt around its bottom opening to effect a sealed joint to the adapter spool. The skirt is raised and lowered by means of motor-driven screw jacks. Inflatable rubber seals provide a gas seal between skirt and cask, while O-rings provide sealing between skirt and adapter spool. The cask is also equipped with a shield gate that covers the bottom opening to protect personnel from excessive radiation when used control rods are transferred to storage. The gate is motor-actuated and equipped with travel stop limit switches, which also function as interlocks to prevent lowering the hoist unless the gate is open or to prevent raising the skirt unless the gate is closed. Built-in, pneumatically actuated snubbers in the walls of the cask steady the contents during acceleration and deceleration of the carriage. Limit switches prevent moving the carriage unless the snubbers are actuated, and also prevent operation of the hoist unless the snubbers are retracted.

A control panel mounted on the carriage serves all cask functions and the carriage travel. The two motors driving the carriage are 5 hp for the high-speed, and 1/2 hp for the slow speed. A brake that is employed on the carriage is always engaged when no electric power is supplied to the motors. When the carriage controls are in use, interlocks prevent operation of the shield gate, shield skirt, and hoist. The auxiliary cask is also interlocked with the refueling bridge to ensure that the latter cannot be moved while the cask skirt is secured to a reactor nozzle. The controls are also coupled with the reactor isolation valve to permit, by a key switch, the control of this valve from the auxiliary transfer cask control panel.

A hot service facility is provided in the reactor service building. In this facility service, repair, and decontamination work can be performed on components that might be too active to handle without the protection of shielding. The rail system that serves the areas between the reactor and the reactor service buildings also serves this facility. Hence, the two transfer casks and the reactor service building crane may be used in

conjunction with this facility. Approximately 60 different plant items might possibly require servicing or decontamination in a hot service facility at some time, and about one fourth of these will probably require such attention. In the latter category are such items as the control-rod drives and rod guide tubes, the helium circulators, and the fuel transfer machine. Because of the capabilities of the facility, fewer spares would need to be stocked on-site than would otherwise be the case.

The hot service facility contains two work areas, one for relatively short items, such as helium circulators, the other for longer items, such as the control-rod drives and the fuel transfer machine. The first area is serviced by a remotely operated manipulator for the disassembly and handling of shorter components. The second area is serviced by an elevator that straddles longer components. Remotely controlled decontamination sprays can be directed at any part of the components by spray nozzles attached to the elevator. At the bottom of the elevator well provisions have been made for water to a depth of 7 ft, which will provide shielding while handling hot items. This handling for disassembly is done with long-handled tools with the operator on the elevator. A small pool is provided in the short area for the decontamination of smaller parts. Otherwise, items are initially decontaminated in the tanks provided before being placed in the hot service facility for repair or further disassembly.

A 2-ton overhead crane is provided in the facility for internal handling of components. The facility is designed with integral shielding limiting any exposure to operating personnel in a control room to a maximum of 2-1/2 mr/hr.

In addition to the equipment described above for fuel handling operations, a series of auxiliary tools and facilities are provided. The latter include storage wells in the floor of the reactor service building for storing the fuel transfer machine, the reactor viewing device, and control-rod drives. Stands are provided for storing isolation valves and adapter spools, in order to avoid damage to their lower sealing surfaces when they are not in use.

The auxiliary tools are provided essentially for emergency handling operations within the reactor. The tools are basically long-reach tools, designed to retrieve, for example, pieces of broken fuel elements and guide tubes. Others are capable of retrieving smaller objects which may have fallen on the grid plate or of removing and inserting fuel element standoff pins. Should a control rod become disconnected from its drive, a tool will cut the loose end of the rod cable before the rod is removed. In all of these salvage operations, the visual aid provided by the reactor viewing device is employed.

Two 115-ton steel casks, each with a 60-element capacity, are used for fuel element shipping. The elements are shipped in a dry, uncanned condition on 6-axle flatbed railroad cars.

Following the loading of the storage vault with spent elements, a three month period is allowed for decay of afterheat and radiation before the shipment of spent fuel to a reprocessing plant. This allows an annual fuel shipping period of up to nine months. Since the average number of elements removed from the reactor core in a year is 915, the total number of round trips per cask to the reprocessing plant is an average of eight per year, which permits about a month per roundtrip.

When the fuel shipping cask is being loaded, a railroad car with cask is rolled into the area below the service building. Hydraulic cylinders raise the cask to a vertical position directly under the designated opening in the service building floor. This scheme in which the shipping cask is mounted in a vertical position serviceable by the fuel transfer cask has the advantage of readily permitting utilization of existing equipment such as the fuel storage vault isolation valve and the plug removal cask. The valve is used to assure seal-tight access to the cask. The plug removal cask also makes a seal-tight connection with the shipping cask via the vault isolation valve.

3.4 PRESTRESSED CONCRETE REACTOR STRUCTURE (REACTOR PRESSURE VESSEL)

3.4.1. General Terminology

Certain terms are sufficiently important to warrant their definition as they apply to this report. The terms are:

1. Prestressing: A technique of design and construction whereby strains are induced in an object prior to its being strained in service. The prestressed strains and stresses induced are essentially equal and opposite to those caused by service loading and counteract them in a manner that is desirable from a design and service standpoint.
2. Tendon: An assemblage used to induce prestressing. The assemblage most often consists of high-strength steel wires, rods, or strands and any permanently installed accessories that are necessary to the prestressing operation.
3. Post-tensioning: A technique of prestressing in which the forces needed to tension the tendons are exerted against the object to be prestressed after the object has been constructed.
4. Prestressed concrete reactor structure (PCRS): A prestressed concrete structure designed to enclose a nuclear reactor and safely resist the loadings imposed on it. Technically this might be termed a pressure vessel; however, some of the underlying design philosophies differ sufficiently from those used in conventional pressure-vessel design that it is desirable to differentiate between the two.
5. Prestressed concrete: Concrete on which the prestressing operation has been performed.

3.4.2. Physical Description

The exterior of the reference PCRS is a right hexagonal prism approximately 86 ft across the flats and 86 ft high (see Fig. 3.13). The hexagonal shape was chosen in order to simplify anchorage of the prestressing tendons.

NOTES -

1. ONE LAYER CONSISTS OF ALL THOSE TENDONS WHICH ARE PARALLEL TO EACH OTHER, & LOCATED IN THE SAME HORIZ. PLANE.
2. ONE TENDON CONSISTS OF SIX 1/8" DIA. GALVANIZED HIGH-STRENGTH STRANDS ENCLOSED IN ONE WELDED CARBON STEEL RECTANGULAR TUBE 3/4" x 5/8" INSIDE x 1/2" WALL, PLUS ALL END FITTINGS.
3. ONE TENDON CONSISTS OF ONE 1/8" DIA. GALVANIZED HIGH-STRENGTH STRAND ENCLOSED IN ONE 2" SCHEDULE 10 STEEL PIPE, PLUS ALL END FITTINGS.
4. PATTERN OF TENDONS IN BOTTOM OF VESSEL TO BE SAME AS PATTERN FOR TOP, AS SHOWN IN VIEW A-A.
5. ONE SET CONSISTS OF ALL TENDONS IN EACH OF GROUPS ONE THROUGH SIX.
6. LOCATIONS AND NUMBER OF TENDONS ARE TO ILLUSTRATE FORCES AND NOT INTENDED TO LIMIT TYPE OF TENDONS.
7. EACH STRAND SHALL HAVE A GUARANTEED STRENGTH EQUAL TO 136,000 LBS. @ 60% OF ULTIMATE.
8. CONCRETE WILL HAVE A MINIMUM ULTIMATE COMPRESSIVE STRENGTH OF 5500 PSI @ 28 DAYS.
9. FOUR 2 1/4" I.D. x 1/2" WALL AND FOUR 22" I.D. x 1/2" WALL A-155 STEEL PENETRATIONS ARE REQUIRED AT EACH OF SIX PLACES.
10. THE INSIDE STEEL LINER AND ALL PENETRATIONS SHOWN ARE TO BE WRAPPED WITH THE EQUIVALENT OF 1/2" SCHEDULE 80 A-155 STEEL PIPE COOLING COILS AT 5" PITCH.
11. FOR ESTIMATING PURPOSES, REINFORCING STEEL SHALL BE EQUAL TO .75% OF THE GROSS CROSS-SECTIONAL AREA, DISTRIBUTED EQUALLY AT EXTERIOR AND INTERIOR FACES.

Fig. 3.13--Prestressed concrete vessel

The interior of the PCRS is approximately a right circular cylinder, 56 ft 6 in. in diameter and approximately 40 ft high. This shape was chosen because it appeared to be the best compromise between structural requirements and system requirements for the criteria under which the reference plant designs were made. The dimensions referred to here and shown on Fig. 3.13 are those used for the preliminary design and cost of a PCRS.

From the inside out the PCRS consists of (see Figs. 3.13 and 3.14) (1) gas space reflective stainless steel insulation to reduce the heat transmission from the primary coolant; (2) a mild steel concrete liner to act as a gastight membrane, to aid in the removal of the heat transmitted through the insulation, and to act as a form for the concrete; (3) cooling coils through which water will be circulated to remove the heat transmitted through the insulation and liner; (4) concrete (placed in direct contact with the liner and cooling coils) which serves as a biological shield against radiation, which resists the tendon forces when the structure is not under service loads, and which provides a means of transmitting service loads to the tendons and assists in retaining the integrity of the mild steel liner under service loadings.

Embedded in the concrete are the prestressing tendons and conventional reinforcing steel. The conventional reinforcing steel and the tendons are arranged vertically and horizontally. They are located and installed so as to avoid conflict with either the mild steel liner of the main cavity or the containment penetrations. The strands of the tendons are enclosed in steel encasements to reduce friction and allow inspection.

The PCRS is located within a metal leak collector that is located within a fairly conventional-appearing building for reasons other than those established by PCRS requirements.

Through the top of the PCRS are access penetrations for fuel handling and control-rod-drive mechanisms, and also larger penetrations for access to the primary coolant circulators and steam generator bundles. Through the sides of the PCRS are penetrations for steam and feedwater piping. Other penetrations include those for pressurizing and depressurizing the structure and for instrumentation and control lines.

The insulation tentatively selected for the reference design is shown in cross section on Fig. 3.14. The insulation is of the gas-space reflective type. It consists alternately of sheets of thin metal and stagnated layers of helium. The thin metal sheets provide opacity to radiant heat energy and also confine the helium sufficiently to minimize natural and forced convection. These gas spaces, in addition to minimum metal area in the spacing devices, reduce the convected and conducted components of heat energy.

The type, thickness, and configuration of insulation is a subject for further investigation, since the insulation is one of the high cost items for the PCRS. In addition to factors relating to primary system cleanliness and insulation durability (system requirements) there are economic factors of insulation cost as related to heat rejection cost. The reference insulation departs from some of the commercially available insulations of the same type in that the gas spaces are wider in order to eliminate as many of the high-cost foils as possible yet achieve adequate stagnation.

The steel liner and cooling coils shown in partial detail on Fig. 3.14 were chosen as a very rough approximation of optimum economics. In choosing the thickness, very little consideration was given to liner stresses, because those stresses are primarily controlled by the concrete. (Some consideration was given to keeping the liner thick to reduce construction problems.) Since the liner serves as a heat collector and conductor between the cold face of the insulation and the heat removing fluid, the liner will be relatively thick to allow use of a reasonably large coil spacing when this is desirable to reduce insulation costs. Conversely, if lower cost insulations are available, the liner thickness will tend to be thinner, because economics may dictate less heat rejection.

Anchorage of the liner will be accomplished in part by the cooling coils and partly by anchors extending farther into the concrete. At first glance, it may appear that extensive anchorage is necessary to prevent separation of the liner from the concrete. Since the gas pressure will tend to force the metal into contact with the concrete and since a 0-psig gas pressure gives rise to a condition where the tendons compress the concrete and liner to a smaller size than when pressurized, local bowing is not, in general, a problem.

The cooling coils serve as the means of distributing the cooling fluid. The wall thickness of the pipe is partly controlled by the area requirements for the heat-conducting path from the liner. The diameter was chosen to give close to the desired surface area-volume-velocity-coil spacing values as determined by the very rough insulation-heat rejection optimization. To minimize construction and operation problems the flow patterns are as direct as possible; and, for system maintenance during operation, two duplicate sets of coils are furnished. Eight cooling zones have been established, one each for the six steam generator modules, one for the liner area adjacent to the plenum over the core, and one for the liner area adjacent to the plenum under the core. This allows one or more steam generator modules to be shut down and liner cooling in that area to be reduced to prevent local overcooling. Mixing of the coolant in the plenums above and below the core will tend to stabilize temperatures in the liner adjacent to the plenums and thus require large cooling changes only for operating and nonoperating reactor conditions.

Fig. 3. 14--Cooling coil and insulation

Coolant flow directions have been chosen to distribute the coolant adjacent to hotter interior areas and collect adjacent to cooler areas so as to minimize insulation zoning. Equality of flow in the individual coil branches will be achieved by matching equivalent lengths (using orificing if necessary) and keeping the pressure losses due to flow high.

The penetrations into the interior of the vessel pose some problems as they do in conventional steel pressure vessels. One problem that is of more importance on a PCRS than on a steel pressure vessel is that of maintaining an integrity equivalent to that anticipated for the main body of the PCRS. Because the holes cannot provide the dual resistance to leakage provided by the liner and concrete, the penetrations for the reference vessel are doubly contained to prevent gross leakage from the primary system in case the liner should rupture at a point between the concrete and its juncture with pipe and header penetrations.

There are many types of penetrations into the PCRS. The most difficult ones to cope with are those through which steam generator headers pass. In general, they will consist of thermal sleeves to allow a temperature and heat transmission reduction to the gas-tight seal with the PCRS liner. In parallel with this primary seal will be one or more seals designed to allow movement along the header axis, yet reduction in or prevention of gas flow along the header in case of a primary seal rupture. To guard against the assumed hazard of steam header rupture inside the two seals, stops will be provided that resist the thrust of the header and prevent concurrent failure of the two seals. Because of the high steam pressure involved, it appears that controlled leakage from the outer seal may be necessary to prevent the high-pressure steam from rupturing the inner and outer seals concurrently.

Owing to the multiplicity of temperatures, pressures, and header sizes involved in the steam cycles investigated, more work must be done to select the method of providing the backup containment.

As mentioned in Section 4.1., the possibility of providing a conventional, large, lower-pressure secondary containment has been briefly examined in order to assess the probable effect on cost. If this more conventional approach to secondary containment is finally required, rather than the reference design, only the gas-tight pressure seal to the PCRS liner would be provided and the secondary seal would be provided by the more conventional secondary containment.

3.4.3. General Load Description

The first major loads that the structure will be subjected to are the prestressing forces induced by the tendons during post-tensioning. These

forces will induce strains (and consequent stresses) in the concrete and mild steel liner that will be opposite to the strains caused by service loads.

For the reference designs the prestressing forces are considerably greater than the forces created by service loads. This excess of prestressing force in conjunction with the strength provided by the conventional reinforcing steel will provide for loads in excess of those anticipated during normal operation.

The conventional reinforcing steel reinforces the concrete against small local tensile discontinuity stresses caused by the applied or induced forces as well as providing reserve strength for the structure.

During the prestressing operation the high-strength wires, or strands, of the tendons are tensioned, thus inducing high stresses in them. Within a short time relaxation occurs in the tendons which reduces these stresses. In addition, the concrete yields (owing to its normal characteristics), thus further reducing the stresses in the tendons and concrete. The residual prestressing forces are those which are designed to resist service loads and loads in excess of service loads.

The second major load that the structure will be subjected to is a pressure load. This pressure load will produce strains opposite and almost equal to the residual strains induced by the prestressing. With almost equal and opposite strains cancelling each other, the residual stresses in the concrete and the mild steel liner will be low during pressurization.

The concrete and mild steel liner stresses are, in fact, lower when the structure is pressurized than when it is depressurized. Since the structure will usually be pressurized, the usual concrete and liner stresses will be low. In fact, the reference designs are such that small compressive stresses will exist in the concrete and liner at design pressure.

Reserve strength against accidental overpressurization is provided by the residual compression plus the tension capabilities of the conventional reinforcing steel and the mild steel liner and the reserve tension capacity of the tendons. The reserve capacity of the tendons is large, because of tension losses subsequent to tensioning and owing to the fact that this PCRS concept requires many tendons (in excess of those required to provide static equilibrium) to control concrete discontinuity stresses at the cylinder-to-head junction and at the holes in the top head and cylinder walls.

Minor loads that the structure will resist are its own weight, stresses induced by differential expansion as a result of temperature differences, and equipment loads such as the reactor. These loads are minor when compared to the two major loads of the tendons and pressure.

3.4.4. Specific Design Criteria and Methods of Analysis

For the reference design the operating pressure is 450 psia with a design pressure of 518 psig. The maximum design temperature of the concrete is 150°F.

The analytical model for stress analysis consisted of a circular cylinder attached at the top and bottom to elastic rings of the same thickness as the cylinder walls. At the bottom the elastic ring was considered to be attached to a flat plate which formed the bottom closure to the cylinder. At the top the elastic ring was attached to cantilevers projecting radially inward between the steam generator bundle access holes. These cantilevers were attached to another ring that was attached to a grid plate (equivalent to a tube sheet) that simulated the area where the fuel-handling penetrations are. The closures for the penetrations, the cantilevers, the ring, and the grid plate form the top closure to the cylinder.

A summary of general stress analysis methods is presented here. The interior dimensions are set by system requirements. The exterior dimensions were selected by conventional thin-shell formulae because thick-shell formulae were not sufficiently developed for use. The thickness of the cylinder was selected assuming a fully restrained condition at each end. The bottom plate thickness was chosen assuming a simply supported circular plate. The thickness of the top head was increased approximately 25% over the bottom to compensate for the hole effect. The limiting stress for the cylinder and bottom plate was assumed as 2800 psi, and for the thickness determination the material was considered to satisfy the usual basic material assumptions.

As hole sizes and locations developed, the structure was subdivided into the various segments previously described, and the detailed stress analysis was made using the classic discontinuity methods, and, thus, the pressure stresses without tendons were determined. Again using discontinuity methods and dividing the post-tensioning tendons into six general areas of application, stresses for unit tendon loads were computed at control points. Assuming the validity of the principles of super-position, influence equations were derived using pressure stresses and stresses owing to unit tendon loads. These equations were expressed as inequalities in matrix form with stress limits of 2800 psi to 0 psi compression.

The superimposed stresses due to pressure and tendons were allowed to assume the limiting values, or any in-between value, at all computed control points with varying tendon loads. The matrix of inequalities was derived in such a way that the internal stresses, under the combined action of the internal pressure and tendon loads, would vary within limiting values when the tendon forces were constant and the internal pressure was 0 or 518 psig. Thus, the stress limits were satisfied with the PCRS pressurized or unpressurized. This system of inequalities had no unique solution, and further criteria were supplied by (1) mathematically expressing the physical limitations on the number of tendons in the areas of application (such as the geometrical limitation of the available concrete cross-sectional area) and (2) requiring the computed solution to be accomplished with a minimum length of post-tensioning tendons.

By using linear programming techniques, a combination of tendon loads was arrived at which gave stress values in the concrete between 2800 and 0 psi compression with minimum tendon length when the stresses were caused by an internal pressure of 0 or 518 psig and constant tendon loads.

This method of stress analysis produced a design that is aimed at controlling the discontinuity and membrane stresses as predicted by thin-wall formulae and, thus, the total number and location of tendons, as well as wall thickness, is dependent on whether the actual structure sufficiently satisfies the assumptions on which the formulae derivations are based. From the time the cylinder and head thicknesses were selected, it was recognized that this structure does not in all respects satisfy those assumptions. The manner in which the assumptions were departed from is considered to result in a conservative design. This belief is supported by evaluation of reports on European model testing. It is further supported by the fact that the total static equilibrium forces exerted by the tendons are approximately twice the forces created by the design pressure on planes containing the cylinder axis, or perpendicular thereto. Up to design pressure the bending stresses are generally compressive, and the average stress is compressive owing to the excess of tendons. If the pressure should rise above design, bending stresses will tend to become tensile, but average stresses will remain compressive until the internal pressure is approximately two or more times design. In this higher range of pressures, tensile bending stresses are resisted by conventional reinforcing. Further increase in internal pressure will make average as well as bending stresses tensile. Since concrete can resist very little tension, the tensile forces that resist the extreme over-pressure are furnished by the liner, conventional reinforcing, and the

prestressing tendons. The prestressing tendons will always share in resisting tension forces, until they break or yield, or until their anchorage fails, because the ends of the tendons cannot be displaced relative to the center without inducing tension by direct or catenary action.

3.5. STEAM GENERATORS

The six steam generator sections provide the means to convert the feedwater from the high-pressure feedwater heaters into superheated steam for the high-pressure turbine, and to reheat the steam for the intermediate-pressure turbine. These steam generators are located in the annular space between the outside of the side reflectors and the inside of the PCRS. Each section consists of four tube bundles, two identical bundles for raising the water temperature from 547° to 835° F, one to superheat the steam from 835° to 1050° F, and one to reheat the steam from 660° to 1050° F. The helium coolant stream is divided at the generator inlet such that one half passes through the superheater and one boiler bundle in series and the other half passes through the reheater and the other boiler bundle in series. The two gas streams are mixed before entering the helium compressor. Figure 3.15 shows the temperatures in the steam generator sections at full load.

The superheater and reheater bundles are arranged for parallel-crossflow. This arrangement requires some additional heating surface over a counter-crossflow arrangement, but results in considerably lower maximum tube-wall temperatures. Figure 3.16 shows the arrangement of the tube bundles.

The superheater tubes, which are bent in the form of a U, connect at the top to round, horizontal headers which extend through the wall of the PCRS. The tubes, which are 5/8 in. O.D. by 0.134 in. minimum wall on a 13/16 in. triangular pitch, are made of austenitic steel and are designed for 3800 psig. The tubes are butt welded to stub tubes which in turn are welded to the headers on the outside. This method permits radiography of the butt weld if desired and permits the header hole to be the inside diameter of the tube rather than outside diameter, which results in higher header ligament efficiencies. The tubes are spread around the header to gain access for welding and to further increase the header ligament efficiency. (For typical details, refer to Fig. 3.19.)

The reheater bundle is similar in design to the superheater, except that the tubes are 1 in. O.D. by 0.109 in. minimum wall on a 1-11/32 in. triangular pitch; they are also made of austenitic steel. Larger diameter tubes are used in the reheater to reduce the pressure drop, since reheater pressure losses have a stronger influence on cycle efficiency than do superheater pressure losses. The tubes are designed for an external pressure of 533 psi, which is the governing condition.

The economizer-evaporator tubes are formed into serpentine panels with eight passes. The tubes, which are 3/4 in. O.D. by 0.134 in. minimum wall on a 1 in. triangular pitch, are made of 1-1/4 Cr-1/2 Mo-Si

Fig. 3. 15--Temperatures in steam generator sections at full load

Fig. 3.16--Tube bundle arrangement

material. The economizer-evaporator tubes are connected to the headers in the same manner as the superheater and reheater tubes.

The tubes for all bundles are spaced and supported by tube support plates which consist of a series of scalloped bars pinned together to form a solid plate. Because of the tube bends required to enter the headers around their periphery, the tube support plates must be built up with these bars as each row of tubes is inserted into the header. The gas is channeled through the bundle section by means of boiler side plates, bottom plate, and the top baffle plate. The header section is also boxed in to prevent excessive leakage through the top baffle plate. Since the steam headers are attached to the concrete pressure vessel through thermal sleeves and the boiler support structure is fixed radially at the center of the circulator, there will be differential expansion between the tube supports and the boiler side plates. This movement is accommodated by allowing the tube support plates to slide on round bars which tie the boiler side plates together. These bars also act as stiffeners for the side plates. As the unit heats up and the tube bank moves relative to the side plates, baffles prevent a gas leak patch from developing around the bundles. The boiler side plates are insulated with 1-1/2 in. of metallic insulation to reduce the heat loss from the bundle and to permit the use of carbon steel for these plates. In order to provide a compact arrangement, all tube bundles are designed to the same width and the same length.

Maintenance for the steam generator section primarily involves procedures and equipment for handling leaky tubes. The most likely source of a leak from the steam side to the helium side of the system is a defective tube that may be leaking either at the tube-to-header weld or along the tube.

The round, straight tube headers extending horizontally through the concrete PCRS wall provide access for plugging leaky tubes. The steam lines are connected to the headers through Tees which are fitted with a blind flange on the run of the Tee. The flange is removed for insertion of tube plugging equipment. Two tube plugging machines are provided to permit leak testing and plugging of both ends of a tube.

Instrumentation is provided in the primary system to detect and locate leaks between the primary and secondary coolants and to locate and shut down the loop which contains the leak. To repair the leak the plant is shut down and the primary coolant loop depressurized to slightly below atmospheric pressure. The header blind flanges for one tube bundle in the defective steam generator are removed. The two tube plugging machines are then inserted into the headers and sealed in place. Tube service operations are performed by remote control.

The tube plugging machine has a telescoping column carrying a rotatable turret which can be traversed the length of the tube connection area of the header. The turret carries tools for leak testing individual tubes, reaming, plug insertion, and plug welding. The two machines are operated simultaneously in leak testing each tube by capping both ends and drawing a gas sample from inside the tube into a helium leak detector. When a leaky tube is discovered, each of the additional tools on the turret is brought into position to ream, plug, and seal weld the tube at both ends. The plugged tube is finally leak tested to determine weld integrity at both ends.

If major repair of a tube bundle should ever be required, the tube bundle can be removed from the PCRS. The over-all dimensions of each bundle, including a portion of the headers, are such that it will pass through the circular opening in the PCRS top head containing the helium circulator. The tubes, headers, tube supports, boiler side plates, bottom plate, and box section around the header form the removable package. In the installed position this package is supported from the PCRS by means of the boiler side plate structure. The cantilevered end of the header and the tubes are supported by the boiler side plate structure through a slide block at the free end of the header.

For removal of the steam generator sections, the circulator, diffuser, and associated internal ducting are first removed. The steam headers are then cut, utilizing the blind flange opening of the header for access of cutting equipment. The first section to come out would be one of the evaporator-economizer bundle, since they are closest to the circulator opening. This bundle is moved laterally until it is directly under the circulator opening. At that point it may be moved vertically upward through the circulator opening into a removal cask. In a similar manner the remaining three bundles may be removed.

Assembly of a new or repaired tube bundle essentially reverses this procedure, with either the superheater or reheater bundle going in first. Jacking arrangements for lining up the headers prior to welding would be required. The header-to-header weld would be performed from the inside, using the blind flange at the end of the header for access. Inspection of the finished weld would be by radiography if radiation levels permit good evaluation of the film, or by an ultrasonic technique utilizing a transducer on the inside of the header. Previous work done by General Atomic has shown that either method appears feasible, although additional development work will be required. It is recognized that tube bundle removal would be difficult and time consuming; however, the plant is laid out in such a manner as to make this possible. Table 3.7 shows the pertinent features of this unit. Details of a typical tube bundle are shown on Figs. 3.17, 3.18 and 3.19.

Table 3.7

STEAM GENERATOR DATA

(Quantities Shown Are For One Bundle)

	Economizer-Evaporator	Superheater	Reheater
Number of bundles	12	6	6
Helium flow, lb/hr	690,000	690,000	690,000
Water/Steam flow, lb/hr	586,000	1,172,000	1,037,000
Number of helium passes	1	1	1
Number of tube passes	8	2	2
Helium pressure drop, psi	2.1	0.5	0.5
Water/Steam pressure drop, psi	30 + orifice	235	40
Heat exchanged, Btu/hr x 10 ⁶	422	235	235
Effective LMTD, °F	195	373	391
Transfer rate, Btu/hr-ft ² -°F	188	175	135
Transfer area, ft ² (effective)	12,660	3,970	4,840
Tubes:			
Number	375	564	430
Outside diameter, in.	3/4	5/8	1
Minimum wall thickness, in.	0.134	0.134	0.109
Pitch (triangular), in.	15/16	27/32	1-11/32
Headers:			
Inside diameter, in.	10	10	16
Bundle Size:			
Length (incl. headers), ft	33	33	33
Width, in.	58	58	58
Depth, in.	51	20	26
Dry weight, lb	73,400	26,800	28,300

3.6. CIRCULATORS

In a gas-cooled reactor plant where power is produced by steam expanding through a steam turbine in a conventional Rankine cycle, a means for circulating the coolant gas through the primary circuit is required. The primary coolant gas circuit consists of the reactor, steam generator, and the circulator compressor system. While a number of different means exist for providing the work for pumping the gas, this choice is greatly influenced by the nature of the coolant gas, helium, and the flow requirements of the reactor. The volumetric flow and the low molecular weight of helium suggest the use of a high speed axial compressor, the blade speed being limited by the stress level. If a low rotational speed driver such as an electric motor of standard 60-cycle frequency or multistage steam turbine is used, the required high blade speeds and flow can only be accomplished by small blades on large diameter disks or, in aerodynamic terms, high hub-to-tip ratios, which increase the size of the machine, its weight, and cost. On the other hand, low blade speed results in a large number of stages with attending complexity, high weight, and cost.

Two types of circulators have been investigated for use in TARGET plant design--a series-steam-turbine-driven circulator and a gas-turbine-driven turbocirculator. Both types utilize high-speed axial-flow compressors.

The other factors which normally influence the cost of gas circulation adversely lie in the requirements for casings, connecting piping, and expansion joints; these can be eliminated or greatly reduced by integration of the circulator component into the primary circuit. This is only possible if the circulator is of reasonable size, as is the case with a high-speed design.

Based on studies carried out to date (see Section 4.2) the series-steam-turbine-driven circulator has been selected for the reference design. The use of a turbocirculator would result in a higher over-all plant efficiency, approximately 46.5% versus 45.8% and a proportional reduction in fuel cycle cost. While the size of the reactor can be reduced slightly by virtue of the higher plant efficiency, certain parts of the plant, notably the circulators themselves and the internal ducting, are increased in cost as a result of using the turbocirculator instead of the series steam turbine. As a result, there appears to be little if any capital cost advantages to be gained by using the turbocirculator. It might be possible that the plant could be redesigned so as to efficiently utilize a higher pumping power and, as a result, the turbocirculator might show a larger cost savings. However, as the pumping power is increased, the cost of the turbocirculator increases rapidly, and it is unlikely that a large power cost reduction could be realized by using a turbocirculator.

Fig. 3.17--Typical tube support and shroud details (reheater)

Fig. 3.18--Typical tube bundle detail (reheater)

Fig. 3.19--Section through typical
steam header

For these reasons and also because of the high development cost of the turbocirculator, the series-steam-turbine-driven circulator has been selected as the reference design.

Figure 3.20 shows the series-steam-turbine-driven helium circulator in a gas-cooled reactor plant. The steam turbine drive is in tandem or series with the main steam turbogenerator set; thermodynamically it is part of the main steam turbogenerator set although mounted on the circulator shaft. Two stages of the high-pressure element of the turbogenerator have to be removed to allow for this unit to be located in the cold re-heat line. The steam enters the circulator turbine on its way to be reheated in the steam generator, and since the circulator is integrally mounted with the steam generator within the reactor containment, this does not result in any large addition of steam piping or additional penetrations of the containment vessel. It is apparent that the energy expended in driving the compressor had to be produced in the plant, but the thermal equivalent of this energy remains in the primary-secondary gas circuit. There is a loss of work from the main steam turbogenerators and a decrease in thermal efficiency of the plant. However, this loss of net output can be compensated by an increase in plant steam flow and heat. This increase in heat and steam flow is a function of the pumping power required, the thermal efficiency of the plant, and the efficiencies of the series-steam-turbine-driven helium compressor, and it results in a loss of thermal efficiency to the plant. The plant heat balance for the series steam turbine case is shown in Fig. 3.21.

3.6.1. Steam-turbine-driven Helium Circulator

The series-steam-turbine-driven helium circulator is a concept which was initiated by General Atomic and has been studied for a variety of gas-cooled reactor applications prior to the TARGET program. For the TARGET plant the unit is mounted vertically between steam generator elements in an annulus around the core as is shown in Fig. 2.2. Both the turbine and the compressor are overhung from the circulator center housing, with the compressor end in the primary helium containment and the steam turbine end in the secondary containment. The central housing contains the thrust and journal bearings and the oil and gas seals.

The steam flow from the high-pressure element of the main turbogenerator set is divided into six streams, one for each of the six circulators. In each circulator the steam enters the steam turbine inlet casing which contains a combination throttle and bypass valve that is used for speed control of the circulator. The steam expands through the turbine nozzle and through the rotating blade row and decelerated

Fig. 3.20--Series-steam-turbine driven circulator

Fig. 3.21--Heat balance

in the exit diffuser prior to discharge from the casing. The discharge pipe is directly connected (outside the PCRS) to the reheater section of the steam generator associated with the circulator. The steam turbine stage is of the impulse type, with sufficient reaction to prevent flow separation in the rotating blade row.

The accessory systems are located in an air room (a small secondary containment) on the top floor of the reactor building and contain the oil pumps, buffer gas compressors, separator tankage, and the instruments and controls necessary for operation.

The thrust bearings are of the flooded Kingsbury type, tilting pad design with provisions for lifting in starting to prevent wear. The journal bearings will be of fixed or tilting pad design.

Any equipment which may require servicing is duplicated and can be isolated without interruption of circulator operation. The clearances between the rotating elements are sufficiently large that significant wear will not be encountered.

The compressor and turbine aerodynamic design are based on acceptable industrial standards and are well within the present state of the art. The stress levels in the blading, disks, and other components are sufficiently low that these components are capable of better than 100,000 hr of life. The operation of the machine will not excite any resonant frequencies in either the blading or shaft critical speed. The basic design of the machine is such as to permit 40% overspeed without any deleterious effects.

The control of the circulator operation is accomplished by means of a combination throttle and bypass valve at the steam inlet end. The operating characteristic of the circulator is such that at part-load operation of the plant there is an excess of steam which must be bypassed around the circulator drive turbine. However, to allow for individual and independent control of each circulator, the total pressure drop in each of the parallel steam lines between the high-pressure and intermediate-pressure elements of the main steam turbine generators must be equal. This is accomplished by the combination throttle and bypass valve.

Overspeed protection of the circulator is provided in two steps by two devices. The first device is an overspeed limiter which moves the combination throttle and bypass valve. The second is a completely independent system which is part of the plant protective system and which closes a separate steam stop valve located in the reheat line.

A cold helium valve which can be used for loop shutdown is located at the compressor inlet. The valve consists of a cylinder which has helical cam slots machined on its outer surface. The cylinder is guided by rollers mounted on the compressor casing and is actuated by a rotary gas actuator also mounted on the casing. The motion of the cylindrical gate is helical, lowering it into the compressor inlet passage to stop the flow.

It may be possible to improve on the installation of the series steam turbine by reversing the flow through the steam generator, which would eliminate the ducts above the steam generator and save reactor vessel height. However, the mechanical design would be the same.

3.6.2. Circulator Handling Equipment

Handling of the helium circulators for purposes of servicing takes place from the reactor operating floor.

First, the large secondary containment cover over the selected circulator access opening is removed complete with shield plug. Next, the two steam pipes and the various service lines to the circulator are disconnected. The mounting bolts for the circulator unit are then removed, and a transfer cask is placed above the access opening.

The helium circulator transfer cask is essentially cylindrical in shape and open at both ends. At the top a shoulder and latch arrangement permits securing the circulator in the cask so that both components can be handled as a unit. At the bottom end, a heavy door is used to close the cask after the circulator has been drawn up into the cask. Construction materials for the cask are steel and lead, with the lead providing shielding, so that operating personnel will not be subjected to dose rates in excess of 2-1/2 mr/hr under normal working conditions. After removal of the circulator from its access opening, a cover is used to close the opening.

The above features of the cask are not yet firm, since further design studies of the services to be performed in this area need to be performed. They may result in a single cask design, suitable for handling the helium circulators as well as the steam generator tube bundles.

3.6.3. Helium Circulator Servicing

The main components which become radioactively hot in the helium circulator are the compressor rotor and stator, which are exposed to the

helium flow. The important plateout is from La^{140} , which emits 1.6 Mev gamma radiation. This is the daughter of Ba^{140} , which has a half life of 17 days. This radiation determines the necessary for the use of a shielded cask for the transfer of the circulator to a shielded area for decontamination and disassembly. A small amount of activation materials in the bottom end of the circulator also occurs.

A hot service facility is provided for in-plant maintenance of hot items. It consists of a deep well for servicing tall components, such as control-rod-drive assemblies and the fuel transfer machine. Another area is provided for shorter components. The floor of this area has a decontamination pit. Remote manipulators and controls, remotely operated crane, and shielding windows are provided. The top part of the cell is made of stepped concrete slabs which may be removed to permit various components to be lowered into the cell by an overhead crane.

One possible method of handling the circulator for servicing operations is given in the following description. Much further study must be made of the detailed operations required to accomplish such an operation.

The circulator and cask are lowered into the cell as a unit. The bolts attaching the bottom of the cask are removed by direct manual access. Personnel are then removed and the cask lifted from the cell leaving the circulator resting on the cask bottom on the facility floor. A sling will be attached remotely and the circulator lifted and rotated 180° so that the compressor rotor is on top.

The circulator is then remotely disassembled with the manipulators. A self-centering socket is provided at the bottom to accept the nut from the shaft, which connects the steam turbine to the helium rotor. This permits the helium rotor to be removed from the main shaft. The split rings holding the stators, the stators proper, and the end housing are removed in sequence, leaving the two rotors exposed for removal. A special fixture is placed around the rotors which extends to the bottom of the housing and is then fastened to the housing and rotors. This fixture permits the rotors to be removed as a unit with the housing, thus preventing their possible damage. The rotor is then removed to an inspection area in front of the shielding windows, inspected, and placed in a shielded cask.

Personnel may then enter the inside of the facility for the manual removal of the remainder of the circulator and to transfer it to a shop area. Shafts and bearings are removed. Oil lines are flushed. A general inspection, repair, and/or replacement of parts is made. The remainder of the reassembly is performed remotely in the hot service facility if the original stators and rotors are reused; otherwise, the reassembly may be performed

manually. The housing fixture is then removed and the compressor stators installed. The transfer cask is lowered around the circulator remotely, the cask bottom is attached manually, and the circulator and cask are removed from the hot service facility.

Certain design features will require incorporation in the circulator design to facilitate remote disassembly of the unit.

An alternative to the remote disassembly method described above could be to etch the radioactive plating from the circulator rotors and stators by use of decontaminants. The units then could be overhauled by direct manual access. The circulator would be located in the decontamination pit in the hot service facility and flooded with decontaminants to a depth covering the rotors and stators and then etched and rinsed. This method requires less time for unit overhaul, but it would probably damage the rotors sufficiently to prevent their reuse and is therefore recommended only when the rotors are known to have been damaged beyond reuse.

A limited visual inspection of the condition of the last stage compressor and rotor may be made without disassembly. This inspection is made in the hot service facility with the aid of suitable mirrors and inspection devices.

A prototype circulator of the type used here is being developed currently under a development program for another reactor project.

3.7. DUCTING AND PCRS INTERNALS

The following description covers TARGET ducting and internals including the core barrel, upper thermal shield, outer reflector, core plate structure, emergency cooling system, steam generators, circulators, ducts, and valves (see Figs. 3.4 through 3.6 and 3.22).

3.7.1. Core Barrel

A cylindrical steel core barrel 37 ft in diameter and approximately 40 ft long surrounds the core and reflector region and separates it from the outer annular region which houses the ducts, boilers, circulators, etc. The core barrel attaches to the vessel roof and seals pressure through an expansion joint capable of taking approximately 1 in. of radial expansion.

Hangers to the ceiling of the PCRS support the core barrel and its loads, which consist of part of the nonremovable peripheral graphite reflector fastened to the inside diameter of the core barrel. At the bottom, the core barrel attaches to the PCRS floor through another expansion joint which takes about 2 in. of axial expansion in addition to radial expansion. Further investigation into the design of these expansion joints remains to be done.

The core barrel is cooled by diverting a small portion of the core inlet helium between the reflector outside diameter and the core barrel inside diameter similar to the way in which the reflector is cooled. At the top, holes are provided for twelve inlet ducts to the six steam generators. At the bottom, beneath the lower expansion joint, holes for six core inlet ducts 2 ft high are provided around the core barrel circumference. The core barrel is shown in Fig. 3.23 and the inlet ducts are shown in Fig. 3.24.

3.7.2. Upper Thermal Shielding

An upper thermal shield is provided under the ceiling of the vessel directly above the core to protect the steel liner and concrete ceiling from radiation damage. The shielding consists of one layer of 9-in.-thick slabs of graphite, above which is an overlapping layer of 9-in.-thick slabs of borated graphite. Above the borated graphite and below the steel liner is approximately 7-1/2 in. of all-metal insulation to reduce heat loss and to keep the liner and concrete cool. The insulation limits concrete temperature to 150°F and heat loss to about 500 Btu/hr-ft², which is removed by cooling coils located just above the liner. This

Fig. 3.22--Partial layout of circulator and steam generator ducting

Fig. 3.24--Core inlet duct

design permits most of the heat generated in the graphite to return through conduction and convection back into the main helium coolant stream, which flows by the upper shielding before passing to the steam generators.

3.7.3. Outer Reflector

The side reflector, which consists of 3 ft of graphite, protects the PCRS wall and the internals in the annulus against radiation damage and excessive thermal heating and improves the neutron economy by reducing the neutron losses from the core. The inner foot of graphite is made up of three rows of graphite rods the same size as the fuel elements. These inner reflectors may be removed by the refueling machine, just as the fuel elements are. Most of the outer 2 ft of graphite reflector is made of rectangular blocks small enough to be removed through control-rod-drive nozzles. The outermost 4 in. of reflector are borated and fastened to the inside surface of the core barrel. The reflector is cooled by passing approximately 2% of the primary coolant flow through vertical passages in the reflector. To protect the circulators from excessive radiation, the outermost reflector blocks extend above the top of the core in the regions in front of the circulators.

3.7.4. Core Plate Structure

The core plate structure consists of many diamond-shaped solid steel plates 4-in. thick and 5-1/2 ft. on a side, all lying adjacent to each other near the bottom of the reactor. This size and shape results from the pitch layout of the fuel elements and control rods, from the desire to minimize warpage effects, and from the necessity of drilling long horizontal holes throughout the core plate for the removal of purge gas from the fuel elements. The purge gas passes from one plate to an adjacent plate through short connecting plates and finally emerges through take-off pipes near the center of the core plate assembly. Vertical holes in the core plates are provided for main coolant flow passage, fuel element standoff pins, and control-rod emergency shock absorbers.

A 2-in.-thick steel thermal shield to protect the stressed core plate from excessive fast-neutron exposure is supported in place just above the core plate by 2-in. spacers. The core inlet helium flows through holes in the thermal shield, thus ensuring cool operation. The core plate is held up by support structure which in turn rests on the PCRS floor.

3.7.5. Emergency Cooling System

The emergency cooling system consists of six independent emergency cooling units in the annulus. The system is capable of removing stored and decay heat from the reactor so that the temperature of the hot helium leaving the core at no time exceeds 1750°F in the event of main coolant loop failure at full system pressure. The heat exchangers, which operate by natural convection, are located at the top of the annulus to gain maximum cold-leg driving head. Short inlet ducts just above the core connect with the emergency heat exchangers. Long cold ducts leave the heat exchangers and return to a point just below the grid plate where they communicate through the core barrel with the region below the core. These exit ducts contain butterfly valves that are closed during normal operation to prevent the coolant from bypassing the core.

3.7.6. Internals in Annulus

Cooling of the internals within the annulus is accomplished by natural convection of 735°F helium flowing up the inner wall of the annulus and down the cooler outer (insulated) concrete wall, in which are embedded cooling coils for removing some 1.4 Mw of heat. Two of four inches of all-metal insulation are provided on the outer wall. This insulation is sufficient to keep the liner and concrete cool. All ducting, boiler boxes, etc., that contact hot 1500°F gas are insulated on the hot-gas side. This protects the stressed members from the high-temperature gas, limits heat losses, and permits cooling of the internals by natural convection in the annulus.

3.7.7. Component Removability

The design is such that everything inside the core barrel except the core plate structure may be removed through the 18-in. diameter control-rod-drive nozzles. Likewise, all internals within the annulus may be removed through the circulator access holes. First, the heads over the access ports and circulator are removed. Next, the ducts may be disjoined from the boiler boxes so that they can be removed. After the steam headers have been cut, the steam generators, which are hung from rails fastened to the PCRS wall near the top, may be slid to positions below the 8-ft. diameter holes. The steam generator top ducts are removed. The steam generator bundles then may be separated at a joint between them so that they may be removed through the access holes. At this point, access to the dust filters and auxiliaries may be obtained. Also, the inlet hot ducts and the emergency cooling system butterfly valves may be removed if so desired. Thus, removal of internals will be feasible, though by no means elementary.

It is evident from the preceding pages that many problems of thermal expansion, remote removability, and remote replacement, etc., of internal components still require a great deal of further investigation. For example, a rather comprehensive study of expansion joints of the various ducts and headers is in order to optimize workability, reliability, cost and at the same time permit relative ease of remote assembly and disassembly at many of these joints. However, in order to solve many of these problems, finalization of many aspects of the larger items such as steam generators, circulators, steam headers, is a prerequisite.

3.7.8. Assembly Sequence of Reactor Internals

Certain reactor internals are installed during the construction of the PCRS. Other items, which are normally considered replaceable, are installed after the PCRS has been completely fabricated. Initially, the concrete will be poured to approximately the elevation of the top of the PCRS "floor." The floor liner, the vertical cylindrical steel liner, the liner cooling tubes, the tendon-housing tubes, and the reinforcement metal are then installed. After this the cylindrical liner is welded to the floor liner and grout is placed under the latter. The cylindrical portion of the concrete is then poured to the height of the PCRS ceiling. Certain reactor internals are then installed.

The core barrel is lowered into place either as a partial assembly or in strips that are later welded together. One vertical strip of the core barrel is omitted until after the core grid plate has been installed. Next the six core inlet ducts are installed and connected to the bottom of the core barrel, after which the support skirt is installed. The top concrete liner plate, nozzles, and reinforcements are located, the liner plate is welded to the cylindrical liner, and the top concrete is poured after the top liner plate has been very firmly shored up.

When the shoring for the upper concrete has been removed, the remainder of the nonremovable core internals are installed through the 8-ft diameter circulator access ports and then through the core barrel side opening. The core grid plate support, the grid plate, and the top thermal shield and insulation are then installed in sections. The core barrel is completed by adding the vertical section, which had been left off for access to the core area. The core barrel is then jacked vertically and attached to the hanging strap supports, after which the bellows-type seals are attached.

No further access to the inside of the core barrel is required for large components, though access for cleaning and installation of

instrumentation, etc., is available through many openings. All succeeding installation and removal of equipment into the annulus is through the six 8-1/2 ft access ports. The heat exchangers are lowered in sections through the access ports onto tracks and slid sideways into position where the header ducts are connected. The concentric turbine exhaust and compressor ducts and the turbine inlet ducts are installed through the access ports. The last item to be installed is the circulator unit.

The above, while straightforward for major operations, implies a large number of detailed operations that have not yet been closely examined. Such examination must be made early in the program in order to ensure economy of construction time and effort.

3.8. CONTROL RODS AND DRIVES

The reactor is controlled by the selective movement of 242 control rods operated in pairs by 121 control-rod drives. The control rods move vertically within the guide tubes which extend through the reactor core, the outlet plenum above the core, and the nozzles atop the reactor vessel to a point of attachment to the control-rod drives, which are mounted in the nozzles. The control-rod drive, two control rods, two upper guide tubes, and a shield plug constitute an assembly which is removable from the reactor as a unit; the lower guide tubes remain in the core and are removed during refueling by the fuel handling equipment.

3.8.1. Control-Rod Drive

The control-rod drive is a winch-type device which raises and lowers two control rods simultaneously by the winding or unwinding of cables from two contrarotating drums. The mechanism is contained in a cylindrical housing which fits into the reactor vessel nozzle and is suspended therein by means of a top flange resting in a counterbore at the top of the nozzle. Attached to the bottom of the drive housing is a cylindrical shield plug through which pass the upper guide tubes and control rods. The control rods are suspended from the cables which pass over the drums, and the guide tubes are attached to mechanisms by means of which they may be engaged with the lower guide tubes in the reactor core.

The control-rod drive housing is divided into an upper and a lower compartment by a lead filled shield. The upper drive mechanism assembly is comprised of those parts which can and should be made most readily accessible for servicing. This includes the drive motor, primary reduction gearing, scram clutch, retarder release clutch, scram retarder, rod position transmitter, switches, and electrical connectors. These components are built into an assembly that may be unbolted and lifted out of the drive housing without removing the housing from the reactor nozzle. Shielding and seals permit this operation to be performed safely.

Electrical and gas connections are made to the control-rod drive through sealed connectors in the side of the vessel nozzle near the top. A helium purge gas flow is conducted through tubing in the upper drive compartment and delivered to the lower compartment, from which it is exhausted through the cable guides to the reactor upper plenum.

A prototype control rod and drive assembly of the basic type required here is being developed currently under a development program for another reactor project.

3.8.2. Control-rod Drive Operation

During normal operation of the control-rod drive in either the shim or the regulating mode, the control rods are moved out of or into the core by rotation of the drums driven through the magnetically engaged scram clutch. Rod speed varies between 4 and 5 in. per second in the range of travel, the variation resulting from the spiral winding of the cable. The scram retarder rotates, but at a velocity too slow to have any appreciable retarding effect.

When the control rods are at the fully inserted position, the cables are completely unwound from the drums and the guide pulley rocker arms are rotated to a position in which the "rod bottom" limit switches are closed. If a cable breaks or goes slack, the rocker arm is rotated to a position in which the "slack cable" switch is closed. If cable slack occurs as a result of a control rod sticking during insertion, reversal of the drive will cause the switch to reopen, thus differentiating between a stuck rod and a broken cable.

Rod position indication is obtained from an electrical position transmitter geared to the main drive shaft at the output end of the scram clutch. The transmitter is connected to a position readout on the control panel.

When the reactor is scrammed by a signal from the plant protective system or by the operator, the scram clutch is deenergized, which allows the control rods to drop into the core. The design is arranged to limit the rod drop velocity so that the rods can be stopped at the bottom position with a minimum decelerating force. To withdraw the rods requires a reset of the scram circuit, permitting the drive to be operated again in the normal manner after a few minutes.

3.8.3. Upper Guide Tubes

The control rod upper guide tubes, which are suspended from the control rod drive, are of composite construction. The upper portion is made of stainless steel and the lower portion of graphite. The two sections are joined above a circular graphite baffle which blocks hot coolant gas circulation above the outlet plenum into the control-rod thermal shield tube.

The lower graphite section which passes through the hot gas plenum has holes for the discharge of control-rod coolant gas to the plenum. The bottom end slips over the rod guide tube in the core; the joint is designed to permit swiveling.

The upper guide tubes are held down against the core guide tubes by springs at the upper end in the shield, which maintain engagement of the joints under relative vertical movement caused by thermal expansion.

3.8.4. Control Rods

The control rod is a cylinder 3 in. in diameter and approximately 20 ft long. It consists of a stainless steel sheath filled with absorber segments. The absorbers are of boron carbide-filled graphite containing 30 wt-% boron. The end fittings are of open construction to permit coolant gas to flow through the rod in the clearance space between the absorbers and the sheath.

The control-rod sheath has transverse slots throughout the greater part of its length for flexibility and to permit coolant gas to flow from inside to outside, from which it is discharged into the upper plenum through holes in the rod upper guide tube.

3.9. HELIUM PURIFICATION SYSTEM

The helium purification system provides the capability necessary to limit the fission-product activity and the chemical impurities in the main loop. The fission-product activity results from the escape of various isotopes produced in the fuel element. The majority of these are carried by the fuel-element purge stream to the purification system for processing. Others, such as iodine, bromine, tellurium, cesium, strontium, and barium, are held up in the internal fission-product trap. The short-lived gaseous isotopes which produce long-lived plateout daughters are of major interest. These gaseous isotopes are removed in the first section of the purification system. The chemical impurities, aside from those connected with initial system cleanup following outages, arise mainly from water leakage into the main loop and as impurities in other streams fed to the main loop. These impurities are removed in a side stream chemical cleanup section of the purification system. The purification system is also capable of reducing the long-lived gaseous activity in the main loop as is required for maintenance and refueling operations.

The purification system processes helium from three major sources. The major input to this system is the fuel-element purge. This purge stream carries the noble-gas isotopes, which have been released from the fuel compacts into the system. The flow rate is 4000 lb/hr. Because of the uncertainty in the retentivity of fuel particle coatings at the end of fuel life, the design fission-product activity in this stream is conservatively based upon the release of noble gases from uncoated fuel particles. The design and expected release curves are shown in Fig. 3.25. However, a second set of input numbers is also used to indicate the expected levels of activity. These numbers are based upon experimentally observed releases from coated particles in the General Atomic in-pile loop (GAIL). The former numbers are designated as the design release numbers and the latter numbers are designated as expected release numbers.

A simplified flow diagram of the helium purification system, with equipment sizes, is shown in Fig. 3.26. The initial section of the purification system, which accepts the fuel element purge and the intermittent main loop cleanup stream, removes the xenon and krypton parents of long-lived plateout activity and any entrained water. The stream is first cooled from approximately 700° to 120°F. The entrained water is then removed by a mist eliminator and the short-lived krypton and xenon isotopes are removed in 2400-lb charcoal delay beds. This section consists of two identical trains of these three pieces of equipment. These can be operated either in parallel or in series. Normally, however, they would be operated in series. Each of these trains of

Fig. 3.25--Design and expected release curves

equipment is designed to operate satisfactorily by itself (i.e., with the other train off-line due to equipment malfunction) for approximately one year.

Under normal conditions, the stream is split following this section with the majority of the flow (i.e., 3400 lb/hr) being returned to the main loop and the remaining flow being diverted to chemical cleanup sections for further processing. With steam-driven circulators in the main loop, the pressure drop available is not enough to drive the requisite quantity of purge gas through this section of the purification system, and an additional compressor and spare are necessary to return this flow to the suction of the main loop compressors.

The purification system is designed to limit the CO and H₂ concentrations in the main loop to 10 ppm each under normal conditions. The gaseous activity returned to the main loop from the purification system results in a main loop activity of 9 megacuries for design conditions. However, the expected level is 0.6 megacuries. This activity is composed only of xenons and kryptons, which do not have plateout daughters that increase the maintenance problems.

Should it become desirable to remove all of the activity from the purge stream, the purification system would be designed to process the entire purge flow, i.e., 4000 lbs/hr throughout the entire system. The new basis of design would significantly increase the size of the chemical cleanup portions of the purification system. Thus, the cost of the purification system would be increased substantially. In this case, the main loop activity would be only that due to direct release from the fuel elements.

In the abnormal condition of a major steam leak or a planned system depressurization and subsequent activity cleanup, the entire stream from this initial section will be processed through the chemical cleanup portion of the purification system.

The chemical cleanup portion consists of three subsections: the first of these is the dehydrator bed section; the second is the low-temperature section; and the third is the hydrogen removal section. The dehydrator beds accept helium flow from the initial portion of the purification system. The dehydrator beds remove any water vapor or CO₂ that may be in the helium stream by adsorption on molecular sieve beds. These beds are normally run in series, but they are piped in such a manner as to allow the most recently regenerated bed to be the last of the two beds. These beds are regenerated by heating to approximately 550°F, followed by evacuation down to approximately 100 microns of Hg absolute pressure.

Fig. 3.26--Simplified flow diagram
for helium purification system
with equipment sizes

Normally the effluent from these beds goes to the low-temperature portion of the system. However, during a major steam leak the effluent from these beds will be returned directly to the main loop.

The low-temperature portion of the chemical cleanup section removes krypton, xenon, CO, nitrogen, and other trace impurities from the process stream by adsorption on charcoal beds at cryogenic temperatures. The first of these beds removes principally the krypton and xenon isotopes. These two charcoal adsorption beds run at -250°F . The beds themselves are designed similarly to a shell-and-tube heat exchanger, with the tubes packed with charcoal. They are designed to remove 225,000 Btu/hr, this rate being associated with the design release of activity from the fuel elements. The expected heat load on the beds is 18,000 Btu/hr, and the installed refrigeration capacity for the beds is 60,000 Btu/hr, or approximately three times the expected heat-load requirements. (Additional space has been reserved to install the full refrigeration capacity of 225,000 Btu/hr, if this is required. The probability of requiring the full capacity, although not negligible, is very low. Therefore, the major portion of the system has been designed to remove the total design decay heat load, but the refrigeration capacity, which is relatively easy to install as it becomes necessary, is not included initially.) The heat is removed as the heat of vaporization of boiling liquid nitrogen on the shell side of the beds. This vapor is subsequently recondensed in a nitrogen recondenser and recirculated. The nitrogen boils at approximately 230 psig and -260°F .

When one of these beds becomes saturated with krypton, it is taken off-line and the contained activity is allowed to decay for a period of two months. After this, it is regenerated in a manner similar to the dehydrator beds.

Following the direct-cooled beds are two gas-cooled charcoal adsorbent beds running at -305°F . These beds remove any CO, nitrogen, possibly some Kr^{85} , and any trace impurities that may be present that have broken through the previous beds. The beds are designed for a regeneration frequency of one year and are regenerated in a manner similar to the direct-cooled beds and the dehydrator beds.

The two sets of beds are appropriately interspersed with regenerative and liquid-nitrogen cooled exchangers. The exchanger preceding the gas-cooled beds is supplied with liquid nitrogen at -320°F and approximately atmospheric pressure by a nitrogen recondenser. The capacity of the recondenser is 12,000 Btu/hr. This recondenser also supplies liquid nitrogen to various other systems in the plant.

Following this low-temperature portion, the helium stream is processed by a hydrogen diffusion unit to remove any hydrogen that may be present in the process stream. This unit consists of palladium-silver tubes through which the process stream passes. The hydrogen diffuses through the tube walls and is removed by a vacuum pump, which maintains the pressure on the shell side at approximately 10 microns. This process runs at 800^oF and, thus, is preceded by appropriate regenerative exchangers and heaters.

Under normal conditions, the effluent helium is compressed to the appropriate pressure and is delivered to the helium-handling system. During plant depressurization and subsequent activity cleanup of the main loop, this compressor is bypassed and the two transfer compressors in the helium handling system are used to supply driving force.

3. 10. HELIUM STORAGE SYSTEM

The helium storage system serves two primary purposes: first, it provides storage capacity for the plant helium inventory, and second, it provides a low-pressure collection point for instrument effluents, equipment vents, and miscellaneous leakage of primary coolant helium. In addition, the system provides a supply of high-pressure helium for operation of helium-actuated valves in the primary coolant system. A process flow diagram is shown in Fig. 3.27.

A helium storage system consists of helium storage tanks, a low-pressure helium tank, a helium valve actuator supply tank, helium purification units, helium transfer compressors, and low-pressure transfer compressors. The low-pressure transfer compressors are used only for transferring helium from the low-pressure tank to the helium storage tanks. The helium transfer compressors are used to (1) transfer helium from the helium storage tanks to the primary coolant system; (2) transfer the primary coolant system helium inventory to the helium storage tanks; and (3) maintain helium pressure in the helium valve actuator supply tank.

During normal operation, the low-pressure helium streams are collected and passed through a plateout trap and then through a helium purification unit before entering the low-pressure tank. The plateout trap is an uncooled bed of charcoal adsorbent that removes low-volatility impurities from the helium. The purification units consist of liquid-nitrogen-cooled charcoal adsorbent beds, together with regenerative exchangers to minimize liquid-nitrogen consumption. These units adsorb gaseous impurities from the helium. Thus, helium entering the low-pressure tank is substantially free from any radioactive fission products. Two purification units are provided, one of which is on standby, with automatic controls to switch flow to the standby unit should significant radioactivity be detected in the effluent helium. Helium collecting in the low-pressure tank, which is maintained at 12 psia, is continuously transferred to the helium storage tanks by the low-pressure transfer compressors. One of the compressors is in continuous operation with the second machine in standby service. The standby machine is automatically started if the pressure in the low-pressure tank reaches 14 psia.

The helium storage tanks are maintained between 20 psia and 30 psia by intermittent operation of one of the helium transfer compressors. When the pressure in the storage tank reaches 30 psia as a result of helium being transferred from the low-pressure tank, one of the transfer compressors is automatically started, pumping helium from the storage tank back to the

primary coolant system. The compressor automatically shuts down when the pressure in the storage tank falls to 20 psia. Since the compressors are oil-lubricated, an oil removal unit that consists of suitable adsorbent beds is provided in the compressor effluent line to remove entrained oil and/or oil vapors from the compressed helium.

Helium leaving the transfer compressor is also used to maintain the helium-valve-actuator-supply tank within suitable pressure limits. If the pressure in this tank should fall to 600 psia, an alarm is sounded and valves in the compressor discharge line are automatically positioned to divert the compressed helium into the tank. When tank pressure reaches 1000 psia, the valves are automatically returned to their normal positions, thus returning the helium to the primary coolant system. Should the low-tank-pressure alarm sound when there is insufficient helium in the storage tank to start the compressors, the plant operator can manually bypass a small flow of helium from the helium purification system into the storage tanks. The helium-valve-actuator-supply tank has been sized so that recharging it is not a critical operation time-wise. Also, as a backup to this tank, cylinders of high-pressure helium are provided for valve actuation.

When it is desired to store the plant inventory in the helium storage tanks, the following procedure is followed: The primary coolant system is first equalized in pressure with the storage tanks by diverting purified helium from the helium purification system to the storage tanks at a controlled rate. When pressures are equalized the plant will be at about 210 psia. One of the helium transfer compressors is then started, taking suction from the purified helium line and discharging to the helium storage tanks. A pressure controller in the compressor suction line limits the pressure to 30 psia, which is the maximum allowable for the compressor. When plant pressure has been reduced to 14.5 psia, the helium pressure in the storage tank will be approximately 1000 psia. Once the pumpdown operation starts, storage tank pressure will be too high for the low-pressure compressors to discharge to directly. Therefore, with initiation of the equalization phase of the pumpdown operation, the discharge of the low-pressure compressors is automatically switched directly to the suction of the transfer compressors which, as previously mentioned, is maintained at 30 psia maximum.

If the purpose of depressurizing the plant is, for example, to gain access to the primary system for refueling, it is necessary to remove substantially all gaseous fission products present in the primary system helium. This is accomplished by circulating helium from the primary coolant system through the helium purification system and back to the primary system by using the transfer compressors. When plant pressure has been reduced to about 30 psia, a pressure controller is placed in

service which diverts enough helium from the transfer compressor discharge back to the primary coolant system to maintain the desired 30 psia pressure level. In order to decrease the time required for this cleanup operation, the plant operator has the option of operating both transfer compressors to increase the circulation rate through the purification system.

The primary coolant system is repressurized essentially by reversing the depressurization operation. Helium is bled from the storage tanks to the primary system at a controlled rate until plant pressures are equalized. One of the high-pressure transfer compressors is then started, taking suction from the storage tanks and discharging to the primary system, and the storage tank inventory is transferred to the primary system. With one compressor operating, it will take about 12 hr to repressurize the plant.

3.11. TURBINE PLANT

The turbine plant for the reference design is generally similar to a modern fossil-fueled plant. The only special requirements of importance which the reactor plant imposes on the design of the turbine plant are (1) the need for a high degree of reliability in the supply of feedwater flow to the steam generators and (2) the modification of the main turbine required to accommodate the series-steam-turbine drives located in the cold reheat line. Fig. 3.21 is the heat balance diagram, based on maximum design output of the turbine-generator. Table 3.8 gives the main performance parameters for the plant.

The general arrangement for the turbine plant is shown schematically on Fig. 3.1. The main turbine operating floor is 41 ft above grade level. Most of the auxiliary equipment is located in the basement section of the turbine building. The deaerator is mounted on the roof of the turbine building 105 ft above grade. The main control room is located on the side of the turbine building.

Table 3.8

REFERENCE PLANT PERFORMANCE PARAMETERS

Throttle pressure, psig	3500
Throttle temperature, °F	1050
First reheat temperature, °F	1050
Throttle steam flow, 10 ⁶ lb/hr	6,886
Condenser pressure, in. Hg abs	1.0
Turbine arrangement	CCQF 52 in.
Net station output, kw	1,039,700
Net station heat rate, Btu/kw-hr	7451
Net station efficiency, %	45.80

3.11.1. Main Turbine Generator

The turbine-generator is a 3600/1800 rpm cross-compound, four-flow condensing reheat unit, operating in a regenerative feedwater cycle. The turbine is rated at 998,600 kw at 3.5 in. Hg back pressure and 3% makeup. The unit operates at throttle conditions of 3500 psig and 1050°F and a reheat temperature of 1050°F.

A high-pressure section and the first double-flow intermediate-pressure section operate on the 3600 rpm shaft. The second double-flow intermediate

pressure and two double-flow pressure sections operate on the 1800 rpm shaft. The length of the last-stage buckets is 52 in.

Steam is fed through a multivalve inlet to the high-pressure element, and the exhaust from the high-pressure turbine is returned to the reactor plant where it expands through single-stage turbines, which drive the primary coolant circulators, before passing through the reheaters and returning to the intermediate-pressure element of the main turbine. The circulator turbine drives require the removal of one or two stages at the high-pressure-turbine exhaust of an otherwise standard turbine. The first intermediate-pressure section exhausts through cross-overs to the second section of the intermediate-pressure turbine. Each exhaust of the second intermediate-pressure turbine is connected to a double-flow low-pressure element.

Each turbine shaft drives a conductor-cooled synchronous generator having a combined rating of 1,254,400 kva. The output of the generators is three-phase 60-cycle alternating current at 24,000 volts. The generators feed a single unit transformer with a step-up to 345,000 volts.

A 75% capacity boiler feed pump is connected to the 3600 rpm shaft of the main turbine through a connect-disconnect coupling with a shear-pin arrangement. A booster pump upstream of the shaft-driven boiler feed pump is similarly driven by the 1800-rpm turbine shaft. Turbine bypass lines are provided to permit dumping of steam to the condenser prior to turbine startup or following turbine shutdown.

3.11.2. Condensing Equipment

The main turbine exhausts to the main condensers. Two complete, single-pass, divided-water-box, surface-type condensers with a deaerating-type hot well are used. Each condenser section has 265,000 ft² of surface. The condensers are designed to maintain a back pressure of 1.0 in. Hg abs with 53°F circulating water inlet temperature.

A twin-element, two-stage steam-jet air ejector with inter- and after-condenser provides air removal. A vacuum pump evacuates the system during startup.

Circulating water is supplied to the condenser water boxes by four circulating pumps. Total circulating water flow is 704,000 gpm and the developed head of the pumps is 20 ft.

3.11.3. Feedwater System

The feedwater system utilizes seven-stages of regenerative feedwater heating.

Two half-size condensate pumps take suction from the condenser hot well and feed the deaerator through two parallel streams of three low-pressure heaters. All low-pressure heater drains are cascaded back to the condenser, and all low-pressure feedwater heaters are equipped with drain cooling sections.

The deaerator feeds one 75% main shaft-driven boiler feed pump (through its booster pump) and two 12-1/2% motor-driven boiler feed pumps. The three boiler feed pumps discharge through two parallel streams of three high-pressure feedwater heaters to the economizer inlets of the steam generators. The high-pressure heaters are equipped with desuperheating and drain cooling sections. All high-pressure heater drains are cascaded back to the deaerator.

The top heater extraction is taken from the high-pressure-turbine exhaust upstream from the blower turbine. The final feedwater temperature is 547°F. Extraction points for the other high-pressure heaters are from the first intermediate-pressure turbine. The extraction for the deaerator is from the second intermediate-pressure-turbine exhaust. The extractions for the low-pressure heaters are from the low-pressure turbine.

3.11.4. Auxiliary Systems

The plant is provided with conventional service water, domestic water, fire water, instrument and service air, and turbine lubricating oil systems. An auxiliary oil-fired boiler supplies steam for the helium circulator turbines, the house turbine, and the deaerator during plant shutdown. The condenser for the house turbine serves as an auxiliary condenser during plant shutdown. A flash tank located on the high-pressure turbine bypass line is used to generate steam for the circulator steam turbine drives during plant startup, when water is being circulated through the steam generators.

3.11.5. Electrical Systems

Station service power, for normal operation, is obtained from two 50% capacity 24/4.16-kv transformers. The primary connections are

solidly tied to the leads between the generator terminals and other power transformers. Power from the station transformer secondaries is supplied to four main 4160-v buses, which provide power for large motors, 480-v load centers, and general building power and lighting.

A third 50% capacity transformer provides startup and reserve power. This unit is supplied from the 345-kv system, and through switching may be connected to normal or emergency 4160-v buses. This system is on standby status during normal operation.

Continuous power for critical plant auxiliaries is assured through use of a completely self-contained house turbine-generator set. This set supplies full-time power to one 12-1/2% capacity boiler feed pump and other critical loads. The emergency power system is electrically isolated from the normal systems. For startup or emergency situations, it is possible to connect normal and emergency systems together.

3.12. PART-LOAD OPERATION AND CONTROL

The performance and control requirements of a once-through steam generator differ significantly from those of the more conventional drum-type steam generator operating at subcritical pressure. In a drum-type (recirculating) steam generator, the portion of the heat-transfer surface devoted to superheating and the remaining portion devoted to pre-heating and evaporation are specified by design, and during operation heat must be transferred into each of these two regions at the proper rate in order to obtain steam at the desired temperature. With a once-through steam generator, only the total heat input to the unit need be specified in order to obtain the required outlet steam temperature, and the portions of the heat-transfer surface devoted to the different phases are automatically adjusted as required.

In a helium-cooled reactor system, variations in either the helium flow rate or the hot helium temperature (i.e., the helium temperature at the steam generator inlet) will change the heat input to the steam generator. With a once-through steam generator, the value of either one of these two parameters may be arbitrarily specified, and the remaining one adjusted to produce the required heat transfer for a desired steam temperature. In other words, there are many combinations of helium flow and hot helium temperature that will provide a desired steam temperature for a given steam generator design and specified feedwater flow rate and temperature.

Although various combinations of helium flow rate and hot helium temperature may produce the same outlet steam temperature, temperatures at intermediate points in the steam generator as well as the cold helium temperature (at the steam generator outlet) and the hot reheat steam temperature will vary with helium flow and hot helium temperature. For part-load operation, the optimum combination of helium flow rate and hot helium temperature level is one that maintains constant superheater outlet temperature and, at the same time, holds the hot reheat steam temperature reasonably constant and prevents temperatures at intermediate points in the steam generator from exceeding their full-load values.

For the reference design considered for TARGET, optimum part-load performance will require that the hot helium temperature at the steam-generator inlet be decreased approximately 120°F as plant load is decreased from 100% to 25%. The helium flow rate must be reduced roughly in proportion to plant load; this will require independent control of circulator speed.

In addition to maintaining optimum steady-state conditions over the load range, the plant control system must act to minimize fluid temperature and pressure deviations from their setpoints under transient conditions.

With once-through steam generators, steam temperature is normally controlled by adjusting heat input to the steam generator. Because of the low water inventory of the once-through unit, it is essential that the heat input to the steam generator be closely coordinated with the feedwater flow rates. If the plant is to have a load-following capability, both heat input and feedwater flow rate must be matched to the main turbine demand.

In the TARGET system, either helium temperature or helium flow rate could be adjusted in order to produce a change in heat input to the steam generator. In order to produce the fast-acting control over heat input that is required for once-through steam generators, helium flow rate is selected as the controlling variable, because the relatively low mechanical inertia of the circulator permits rapid changes in circulator speed, whereas the high thermal inertia of the reactor core makes rapid changes in helium temperature difficult to obtain.

The TARGET reference design control system is described in Section 3.13.

3.13. CONTROL AND INSTRUMENTATION

In keeping with modern central station design practice, a single central control room will be established for TARGET. All continuous power-plant systems will be operated from and/or monitored by instruments in this room. This will include all instruments necessary to start up or shut down the plant during normal operation. Infrequent routine tasks will be controlled from local control stations. Infrequent, non-routine operations may be similarly controlled.

3.13.1. Power Plant Control Instrumentation

The power plant control instrumentation is governed by three basic parameters:

1. The turbine throttle pressure.
2. The steam temperature at the outlet of the superheaters.
3. The temperature of the primary coolant helium at the outlet of the reactor core.

In accordance with conventional practice for one-through steam generators, the high-pressure-turbine throttle pressure is controlled by varying the feedwater flow to the steam generators. Appropriate controls are provided to maintain equal feedwater flow to the six steam generators. Further analysis may also indicate the desirability of adjusting the feedwater flow through each section of the economizer-evaporator in order to equalize the two economizer-evaporator outlet steam temperatures. Further analysis is required to establish the control requirements, if any, for the reheater.

Preliminary analysis indicates that the reheat steam temperature may also have to be controlled, since it appears that it does not closely follow the superheater outlet temperature on load changes.

Each superheater outlet steam temperature will be kept constant by controlling the helium mass flow to the respective steam generator. The hot helium temperature from the reactor will be controlled by control rod movement as dictated by an automatic neutron flux controller. Programmed variation of the reactor outlet helium temperature in accordance with the turbogenerator load will be required in order to stabilize the evaporator outlet temperature.

The proposed method of controlling the helium mass flow is to vary the steam turbine circulator speed by adjusting the ratio of the steam flow

through the turbine to the steam flow through a bypass. A circulator-turbine inlet throttling valve coupled with a bypass valve or a combination throttle-bypass valve are alternative control devices.

During normal operation, an increased load imposed upon the main turbogenerator will cause the governor valves to open, thus allowing more steam to flow through the turbine. The increase in steam flow will tend to cause the throttle pressure to drop. Upon sensing this pressure drop, the plant controls will increase the feedwater flow to all steam generators in parallel in order to bring the throttle pressure back to the setpoint. The increased steam flow results in a lowering of the superheater outlet steam temperature. The superheater temperature controller will therefore act to increase the helium mass flow, as will an anticipation signal to this controller from feedwater flow. The increased helium flow tends to lower the reactor outlet temperature. To compensate for this, the reactor temperature controller will withdraw control rods from the core, thus increasing the heat output from the core and raising the helium temperature.

A decrease in load with a corresponding increase in throttle pressure will initiate control action opposite to that described above.

Further analysis is required to determine the response of the system: it is expected that anticipatory control signals will be required to initiate corrective control action on system changes and to minimize system upsets and deviations from set point on load changes. In this connection, fast response temperature and pressure sensors will be utilized to reduce delays.

3.13.2. Reactor System Instrumentation

In-Pile Instrumentation. The function of the in-pile instruments is to provide temperature indication of the following:

- Selected core elements
- Selected reflector elements
- Internal structures and concrete

Two types of temperature sensors are utilized: (1) acoustic thermometers capable of measuring temperatures in excess of 2000°F and featuring a lifetime equal to the life of a fuel element, and (2) thermocouples.

It is estimated that 200 instrumented core elements will be sufficient to allow the determination of the axial and radial temperature distribution

within the core for purposes of flux flattening and the detection of flux oscillations due to uneven xenon burnup within the core. Each instrumented core element will contain one acoustic thermometer and one thermocouple. The purpose of the latter is to calibrate the acoustic thermometer, since thermocouple lifetime in the reactor core is expected to be limited.

About one hundred additional thermocouples will be required to perform the remainder of the functions. The sensors, which consist of thermocouples and tubes, will be routed from their assigned in-pile locations to four small pressure vessels underneath the reactor, containing the acoustic transducers. Only electrical leads will be carried to the controlroom. Primary coolant gas from the reactor outwards will be doubly contained.

Nuclear Instrumentation and Sources. The nuclear instrumentation consists of seven nuclear channels, which are grouped as follows:

1. Two startup channels incorporating high-sensitivity boron-lined counters.
2. Two logarithmic power and period channels incorporating two compensated ion chambers.
3. Three linear power channels incorporating three uncompensated ion chambers.

An overlap of one decade will be provided between each group of instruments. A number of sources of strength sufficient to allow the indication of 5 to 10 counts per second at the output of the start up instruments will be incorporated in the core and reflector elements during the initial startup, and in the reflector elements only subsequent to initial plant startup.

3.13.3. Radiation Monitoring Instrumentation

Area Monitors. Permanent gamma monitors will be installed in working areas subject to accidental above-normal radiation from equipment or lines, or from activity released into the building air. These will provide visible and audible warning to operating personnel both in the working areas and in the control room.

Plant Ventilation-System Monitors. Particulate and gas activity detectors connected into the ventilation-system exhausts will monitor for beta and gamma activity inadvertently released to the building air.

Ventilation ducts serving such potentially dangerous areas as the refueling area, the decontamination system area, and the purification system area will be monitored by individual moving-filter particulate and gas monitors. Less critical areas will be monitored by individual fixed-filter particulate monitors and one gas monitor sequentially switched to a number of ducts.

Stack Monitoring. The purpose of the stack monitoring instrumentation is to ensure that the total activity released to the plant environs is below acceptable levels. As presently envisioned, the instrumentation consists of (1) moving-filter particulate monitor based on the detection of β - γ activity and α activity; (2) gas monitor sensitive to β and γ activity; and (3) iodine monitor.

3.13.4. Control-rod Instrumentation

The basic functions of the control-rod instrumentation are (1) to provide a means for the actuation of the control rods, (2) to indicate the position of the control rods in the core, and (3) to indicate the condition of the control-rod suspension cable, i. e. , whether normal or slack.

The rod actuation controls allow three modes of operation:

1. Manual withdrawal or insertion of any one pair of the 121 pairs of control rods.
2. Manual insertion or withdrawal of preselected six-pair groups of rods.
3. Automatic insertion or withdrawal of one or possibly two pairs of regulating rods to control the reactor power level.

The position of the control rods will be indicated by (1) IN and OUT limit lights, (2) eight manually selectable coarse rod-position indicators, (3) one or two coarse rod-position indicators permanently connected to the regulating rod(s), and (4) one precise position indicator ($\pm 1/2$ in.) manually selectable to indicate position of any one of the 121 rod pairs.

Slack in the control-rod suspension cable, possibly indicating a broken cable, is sensed by a switch and indicated by a pilot light.

3.13.5. Plant Protective System

The plant protective system receives signals from the nuclear and power plant critical parameters and initiates three basic actions to protect

the integrity of the plant when these signals exceed preset limits. These actions are:

1. Control-rod withdrawal prohibit.
2. Control-rod scram.
3. Steam-generator loop shutdown.

In order to improve operating reliability without sacrificing safety, a two-out-of-three coincidence system is utilized whenever safety actions lead to rod scram or loop shutdown. The only exception to this system is the period scram, which is bypassed when power reaches 10% of rated.

Preliminary analysis indicates that control-rod-withdrawal prohibit and control-rod-scram actions will be required if neutron flux, reactor period, reactor outlet temperature, reactor pressure, and steam pressures exceed preset limits. Low feedwater flow in individual steam generators and activity release to the steam will result in steam-generator-loop shutdown. The presence of moisture in the helium will occasion both rod scram and steam-generator loop shutdown.

All safety actions are annunciated. Additional trips are also provided to annunciate off-normal conditions before a safety action becomes necessary.

3.13.6. Primary Coolant System

Primary coolant loop pressures are measured at each helium compressor inlet and at the reactor inlet. Differential pressure measurements are taken across the core. The reactor pressure signals are routed to the plant protective system to initiate rod scram on low pressure; in addition, all the measurements mentioned above are used as a guide to compare the relative performance of the six loops.

Resistance temperature detectors are provided in the hot gas ducts for connection to the flux control system. Resistance temperature detectors are utilized for control purposes because of their superior sensitivity and repeatability. Thermocouples are also installed in the ducts for temperature indication, for alarm, and as inputs to the plant protective system.

Sensing devices for the secondary coolant system closely follow conventional power-plant practice. Pressure measurements are made at each

superheater outlet, reheater outlet, and feedwater inlet. Temperature measurements are made at each superheater and reheater outlet, economizer-evaporator outlet, and the economizer-evaporator inlet for system supervision and performance comparison between loops.

Resistance temperature detectors are provided for each reheat and superheater outlet for the steam-temperature controls.

Each primary coolant loop is provided with a moisture detection system to detect a leaking steam generator and shut down the faulty loop before excessive deterioration of the core can occur as a result of the steam-graphite reaction. It is desirable to detect a moisture leak as quickly as possible for two reasons: (1) to reduce damage to the core, and (2) to identify the leaky loop before the ratio of moisture in the leaky loop to that in a normal loop has been downgraded by mixing in the reactor.

Considerations of damage, primary gas recycle time, and loop isolation valve closure time postulate a maximum delay of the order of a few seconds. For a given mass flow rate of sample gas (limited by the maximum activity that can be handled safely and economically), the sample gas delay inside the reactor can be controlled by bypassing sample gas around the compressor section of the circulator. However, the delay from inside the pressure vessel to the outside of the biological shield is dependent only on the length and bore of the sample tube (and the reactor parameters, which are fixed). Present indications are that slightly more than 6 sec may be required to detect a leak.

Moisture detectors capable of being located within the reactor or gas ducts would eliminate the sample transport time delay. The availability of such detectors requires further investigation.

During part-load plant operation (less than 60% rated power), the pressure of the reheat steam drops below the primary coolant pressure. As a result, there is a possibility of helium leakage into the steam lines. Medium sensitivity instruments will be provided for automatic steam-generator loop shutdown, if the steam activity exceed preset limits. In addition, high-sensitivity detectors will monitor the steam for very small leaks.

3.13.7. Helium Purification System

Sufficient instrumentation is installed to make routine operation of the helium purification system automatic.

The nitrogen system contains cryogenic-type instruments and controls as required for automatic operation.

3.14. SHIELDING

Shielding analysis of the TARGET reactor concept has been limited to those problem areas in which the feasibility of the system or key subsystems might be in question. Areas investigated include the fast-neutron flux incident upon the vessel liner, nuclear heating in the side and top regions of the reactor, and biological dose rates external to the reactor.

3.14.1. Nuclear Fluxes and Heating Rates

Neutron and gamma fluxes and heating rates were calculated for the side regions of the TARGET reactor using an extensive series of IBM-7090 codes. Infinite-slab geometry was used, and attenuation by circulators and steam generators was ignored.

The fast-neutron flux ($E > 1$ Mev) in the side liner varies from 6×10^8 to 4×10^8 n/cm²-sec. A 30-year exposure at this flux level will not appreciably affect the NDT of carbon steel at 150°F.* Therefore, no additional fast-neutron shielding is needed at the side.

Nuclear heating rates are shown in Fig. 3.28. The heating rate at the inside surface of the concrete vessel is 42 Btu/ft³-hr, dropping off with a relaxation length of 1/2 ft. The above is equivalent to a heat input to the concrete of 21 Btu/ft²-hr, which is sufficiently low to eliminate the need for a thermal shield at the side of the TARGET reactor.

Fluxes and heating rates were estimated for the top regions of the reactor, assuming a relative power of 0.7 at the top of the core and a 2-ft (effective) top reflector. With no additional shielding, the fast-neutron flux in the top liner would be 1.5×10^{10} n/cm²-sec (too high from the standpoint of radiation damage). Therefore, an 18-in. thick graphite shield (including a layer of borated graphite) was interposed between the top reflector and the concrete liner. This shield also serves to reduce the nuclear heating rate in the concrete to negligible proportions.

The fast-neutron current incident upon the liner via the nozzle openings was also calculated. As a result of this calculation, 6-in. borated-graphite plugs were specified for the nozzles.

*Pellini, W.S., et al, "Analysis of Engineering and Basic Research Aspects of Neutron Embrittlement of Steels," Welding Research Supplement, October 1962, pp. 455-469.

Fig. 3.28--Nuclear heating rates in side regions of reactor

3.14.2 Biological Dose Rates

Biological dose rates outside the vessel can be shown to be negligible, except where ducts or other irregularities occur. The calculations indicated a neutron dose rate of 2000 rem/hr at point 2 ft into the concrete from the inside surface of the vessel, and a gamma dose rate of 1500 rad/hr at the same point. Extrapolation using reasonable values of relaxation lengths shows that both dose rates become negligible at a concrete thickness of 10 ft. Since the minimum side thickness is 14 ft 3 in., it can be concluded that the PCRS fully doubles as a biological shield.

3.15. NUCLEAR PLANT LAYOUT AND LEAK COLLECTOR

The nuclear steam raising section of the power plant is housed in two main buildings, a reactor building and a reactor service building. These are shown in Fig. 3.1. The reactor building is 146 ft by 146 ft by 84 ft high above grade; it is constructed of conventional materials to a high leak standard, but is not intended to resist pressure. It will prevent the escape of activity released from any of the accessory equipment it houses. A separate leak collector fits fairly closely around the reactor pressure vessel. It will be demonstrated by models and analysis that the maximum credible leak rate from the vessel, even under emergency conditions, is negligibly small compared with the capacity of the main transfer compressors to return the gas to helium storage. It will also be demonstrated as described in Section 4.1, that a catastrophic failure of a PCRS is incredible.

Housed in the reactor service building is the refueling equipment and fuel storage vault. Also on the grade floor are the hot/cold change facility, the first aid facility, the health physics control stations, and the decontamination and radiochemical laboratory. At the basement level are the decontamination and radioactive liquid and gaseous waste facilities. In the reactor building basement are the helium purification system, helium storage system, cooling systems, and instrumentation.

3.16. COSTS

An estimate of the total capital cost of the reference design power plant has been made.

The direct cost of the conventional part of the power plant was estimated by Sargent and Lundy, and the nuclear steam supply systems and all indirect costs or general charges were estimated by General Atomic. The PCRS was priced by Sargent and Lundy. The prestressing, which represents approximately 40% of the total PCRS cost, is supported by an estimate furnished by Joseph T. Ryerson and Son, Inc.

The steam-generator and reheater costs are based on vendor estimates, as are the steam-turbine-driven helium-circulator costs.

All material costs are present-day prices, and field construction labor is based on a weighted average wage rate of \$4.10 per hour.

The following items are excluded from the capital cost estimates:

1. Research and Development costs
2. Escalation
3. Fuel and fuel fabrication cost
4. State sales/use taxes
5. Transmission plant (main transformer only is included)

Interest during construction is 9.4% (42 months at 6%). Contingencies are included at 10%. A cost breakdown is given below in Table 3.9..

Table 3.9

POWER PLANT COST BREAKDOWN FOR REFERENCE DESIGN

(Case 3)

<u>System</u>	<u>Cost (\$1000)</u>
10. Reactor system	\$ 17,046
20. Heat transfer system	22,307
30. Steam generator feedwater system	5,648
40. Plant water systems	4,350
50. Turbine generator units	20,811
60. Decont. and radioactive waste systems	641
70. Structures and improvements.	6,762
80. Misc. power plant equipment.	775
90. Electrical equipment	<u>3,039</u>
TOTAL DIRECT COST	\$ 81,379
TOTAL INDIRECT OVERHEAD	<u>41,861</u>
TOTAL CAPITAL COST	<u><u>\$ 123,240</u></u>
Net send out, Mw(e).	1040
\$/kw	118.5

The following is a list of items included in the different accounts:

10. Reactor Systems

11. Reactor vessel and internals	8,781
12. Rods and drives	3,666
13. Fuel handling equipment.	1,418
14. Fuel storage	496
15. Fuel shipping equipment.	714
16. Auxiliary equipment.	501
17. Side reflector.	<u>1,470</u>
	17,046

20. Heat Transfer Systems

21. Primary coolant system (including helium circulator) . .	6,739
22. Secondary coolant system (including steam generator) .	12,761
23. Helium purification system	1,249
24. Helium storage system	775
25. Radiation monitoring, etc.	<u>783</u>
	22,307

30. Steam Generator Feedwater System

31. Feedwater and condensate	4,977
32. Feedwater heater vents and drains	180
33. Water treatment	<u>491</u>
	5,648

40. Plant Water Systems

41. Circulating water system (including condenser)	3,387
42. Service water system	144
44. Reactor building cooling water system	327
45. Purge cooling water system	71
46. Shield cooling water system (incl'd in item 11)	--
47. Screen wash system	230
48. Domestic water system	70
49. Fire water system	<u>121</u>
	4,350

50. Turbine-Generator Units

51. Main turbine generator and auxiliaries	19,621
53. Extraction steam	477
54. Turbine lubrication oil system	70
55. Hydrogen seal oil system	4
56. House turbine generator and condenser	<u>639</u>
	20,811

60.	<u>Decontamination and Radioactive Waste Systems</u>	
61.	Decontamination system	236
62.	Radioactive liquid waste system	207
63.	Radioactive gas waste system	<u>198</u>
		641
70.	<u>Structures and Improvements</u>	
71.	Reactor building	
	Reactor building (only)	1,680
	Secondary containment	195
	Interspace purification	--
	Interspace cooling	--
72.	Reactor service building	842
73.	Turbine building	3,532
78.	Yard improvements	<u>513</u>
		6,762
80.	<u>Miscellaneous Power Plant Equipment</u>	
81.	Cranes and hoisting equipment	2
82.	Instrument and service air	284
83.	Communications system	35
84.	Auxiliary boiler	<u>454</u>
		775
90.	<u>Electrical Equipment</u>	
91.	Switchgear	889
92.	Switchboards	26
93.	Protective equipment	31
94.	Electrical structures	93
95.	Station service equipment	286
96.	Main transformer	1,228
97.	Miscellaneous electrical	<u>486</u>
		3,039

Table 3.10 gives a breakdown of the power cost in mills per kilowatt hour at a fuel life consistent with minimum fuel cycle cost and also at a life consistent with an approximately maximum conversion ratio. Insurance is in an amount as given in the AEC ground rules published in TID-7025, i. e., \$260,000 annually. Operation and maintenance is also according to the same publication. The curve, Fig. 530-2, has been extrapolated to 1000 Mw(e) with an additional margin to cover some increased complexity, primarily in the various nuclear systems. Similarly, the working capital, as in TID-7025, has been based on 60% of the core fabrication cost. These same assumptions apply to Table 4.18 (see Section 4.5). The following assumptions were used here:

Capital charge rate 14%, 80% plant load factor,
 AEC owned fuel,
 GA modified cross sections,
 Retaining fuel,
 Beryllium oxide spine,
 Core fabrication cost $\$22 \times 10^6$.

Table 3.10
 POWER COST BREAKDOWN

	Power Cost (mill/kw-hr)	
	Min. Fuel Cost	Max. Conversion Ratio
Capital charge	2.37	2.37
Insurance	0.04	0.04
Operation and Maintenance	0.20	0.20
Working capital	0.16	0.16
Fuel cycle	<u>0.75</u>	<u>1.10</u>
TOTAL	<u>3.52</u>	<u>3.87</u>
Conversion ratio	0.954	0.994

3.17. REACTOR SAFETY AND HAZARDS

In the process of developing the TARGET reference design, a number of major inherent safety characteristics have been incorporated to cope with postulated emergency situations. A complete evaluation of safety and hazards has not been made for the reference design, of course. At this stage, it was believed necessary to examine the problems only sufficiently to ascertain their nature, and whether there are inherent safety features that would be undesirable or unacceptable.

Safety aspects of the TARGET design are in many ways similar to those of the Peach Bottom plant, and thus, the acceptance of the Peach Bottom design as regards reactor safeguards is to some extent, indicative of the acceptability of the TARGET design. The most significant differences in the designs, neglecting power ratings, are the use of a PCRS with leak collector type containment, and the use of control rods and drives that operate from the top of the reactor rather than from the bottom.

The section on the PCRS (Section 4.1.1) discusses vessel integrity and concludes that a complete and catastrophic failure of the proposed design concept is incredible. This conclusion, of course, would be expected to be proved by model tests and more detailed analysis before final acceptance, and such a program is proposed as part of the TARGET research and development program for the PCRS. Also, in the referenced section, the maximum credible leak rate of the reactor vessel is discussed and some possible leak rate values are given. Both the maximum credible leak rate and the possibility of catastrophic failure aspects are important to the philosophy of the system, and both must be considered together. It should be noted that all penetrations through the vessel are doubly contained and protected so that their integrity is of the same standard as the vessel structure itself.

The maximum credible leak rate is postulated on the basis of the vessel not exceeding a certain maximum permissible emergency pressure. Even though it can probably be shown that it is impossible to fail the vessel in a catastrophic manner by any pressure rise, it might be postulated that certain conceivable incidents might cause pressure increases which, in turn, might cause severe leaks which exceeded the capacity of the leak-collector type of containment. It is therefore necessary to the proposed concept to prevent the vessel pressure under all conceivable circumstances from exceeding some suitable figure, say about 30% over working pressure.

Means for achieving this positive limitation of vessel pressure remain to be explored further taking both safety and economics into account. The obvious solution, namely provision of a multiplicity of safety relief valves venting into storage tanks, may not be economically attractive, even though it is certainly physically feasible. Since the only mechanism available for producing a large increase of vessel pressure seems to be a release of steam from a steam generator into the reactor vessel, it may be more desirable to concentrate on developing a steam/water dump system (see below) or other pressure control system having as high a standard of undoubtable reliability as a safety valve system has.

3.17.1. Steam Leak into Reactor Vessel

One possibility for overpressurizing the reactor vessel is associated with a leak in steam generators discharging steam into the reactor cavity. The severity of such an occurrence is obviously a function of the size of the leak.

One method of minimizing pressurization effects from a leak in any steam generator subsystem is to scram the reactor, automatically isolate the affected loop and dump the contents of the faulty steam generator into a steam/water dump tank via the generator's feedwater inlet as in the Peach Bottom plant. The dump system would be actuated on a signal from moisture monitors. The amount of water entering the vessel, in this case, even with a 100% steam-graphite reaction and with the serious, but incredible, condition in which all main circulators stopped, would give a pressure rise of only a few per cent, which is of no consequence.

3.17.2. Reactor Vessel Leakage

3.17.2.1. Abnormal Leakage. As discussed in the previous sections, means would be provided to positively prevent the vessel from being pressurized above a maximum permissible emergency pressure, which by definition is a value to which it could be tested and at which it would be completely safe.

It is perhaps worth reiterating that leaks at the penetrations have a very low probability of occurring, because the penetrations are doubly contained, with missile barriers between the containments, which prohibits the failure of one containment from causing failure of the other. A penetration of this type, consisting of an inner or primary

containment and an outer or secondary containment, is similar in principle to the primary and secondary containments in the Peach Bottom plant or in another reactor where a simultaneous failure of both primary and secondary containments is considered incredible.

It is therefore difficult to envisage any circumstances leading to the occurrence of a leak, particularly since the whole vessel structure, both liner and concrete, are under compression.

As is discussed in Section 4.1.1, it has been assumed that for no apparent reason a crack occurs in the liner. Calculations for leakage through concrete based on the most pessimistic test results that it has been possible to find, show that even in the event that 10 ft² of liner should be lost, the leakage rate would be only about 5 lb/hr. Because the liner is fastened to the concrete with closely pitched welded anchor attachments, it would seem incredible that even one square foot should be lost.

3.17.2.2. Normal Leakage. During normal operation, the leakage should be no greater than for a steel pressure vessel. In addition, because of the delay time both in the concrete and in the leak collector space, the activity release should consist of essentially nobles gases only and should be low enough for release up the stack. The leak collector space will normally be held at a pressure slightly above atmosphere to prevent inleakage of air, whose nitrogen makes helium recovery difficult.

3.17.2.3. Maximum Credible Leak. It is not really credible that 10 ft² of liner should be lost. Thus, it follows that the maximum credible leak should be less than 5 lb/hr, which is about 8 cfm at the pressure in the leak collector. Under these conditions, one of the main transfer compressors could be used to pump this leakage gas through a purification system and return it to the primary circuit. The other transfer compressor would be available as a standby. Each transfer compressor has a capacity of 470 cfm, so that this leak, though far larger than is credible, is negligible for the compressor to handle. It is clear that a leak not due to gross overpressure can easily be handled.

3.17.3. Emergency Cooling

An important difference between the concrete and steel pressure vessel concepts, arising from the smallness of the maximum credible

leak in the former lies in the fact that higher coolant pressures during emergency cooling can be considered for concrete vessels. With steel vessels, emergency cooling must normally be assumed to operate at pressures little above atmospheric, whereas with the concrete vessel essentially full system pressure can be utilized for such cooling at least during the initial, critical period of high heat removal rate.

Emergency cooling should therefore be less of a problem with concrete vessels. In fact, on the TARGET plant the emergency loops with natural convection are capable of removing both the sensible and decay heat without exceeding an outlet gas temperature from the core of about 1750°F, which is considered acceptable and should not be damaging to the equipment in the reactor cavity.

3.17.4. Reheater Leaks

During normal operating conditions, the reheater steam pressure is above helium pressure, so that a leak will result in steam entering the helium circuit. This is the same situation that exists in the superheater, evaporator, and economizer sections, and therefore creates no new problems. Each turbine-driven circulator obtains its steam from the cold reheat line; thus, steam will always be flowing through a reheater to cool the tubes, unless the circulator is stopped, in which case no hot helium flow occurs. Whenever the circulator is inoperative, a valve in the helium line is closed to prevent backflow. It thus appears impossible for a reheater to be damaged by hot helium owing to a lack of steam flow. Even if the helium valve should fail to close, the back-flowing helium would be cool.

At part load, or under emergency conditions where the reheaters may contain steam pressures lower than helium pressure, a leak may result in radioactive helium entering a reheater. Such a condition would be detected by activity monitors in the reheat steam system, which would automatically activate valving to isolate the particular reheater.

3.17.5. Reactivity Control

Reactivity control is achieved by means of 242 control rods supported by cables and moved by 121 winch-operated drive mechanisms. These rods and their mechanisms are described in Section 3.8. It might be noted that these differ from the Peach Bottom design in that they are

inserted from the top by gravity and are supported on cables in the manner of most other gas cooled reactors. As a consequence of rod insertion from the top, a rod can only fall into the core, not out of it. Furthermore, because of the large number of rods provided the reactivity controlled by one or two rods is very small, being, in fact, only 0.2% Δk for one rod pair. As shown in Section 3.2.3, full withdrawal of one rod pair, even if not followed by scram, results in a mean core temperature rise of only about 150°C.

3.17.6. Safeguards Conclusions

It is believed that there are many inherent safety characteristics associated with the HTGR concept and that the TARGET reference design has utilized and augmented these features. There do not appear to be any postulated accidents that could lead to or cause an unacceptable potential hazard. Further analysis and additional information from testing programs will be necessary to complete the information required for competent final design and construction of the plant. However, on the basis of the preliminary design studies, the TARGET reference design does not appear to have features that would be unacceptable or undesirable from a safeguards aspect.

IV. STUDIES LEADING TO REFERENCE DESIGN

A considerable amount of effort during the past nine months has been directed toward the selection of a reference plant design. The purpose of selecting a reference design at an early date is to have a focal point for future design and development work. Since power cost, conversion ratio, and plant safety are of prime importance in selecting a reference design, preliminary plant designs and cost estimates were carried out for a number of different plants; the results are shown in Table 4.18. These plant designs were selected in such a manner as to allow a comparison of the following features of the plant.

1. Type of primary system pressure containment--Steel vessel versus prestressed concrete vessel.
2. Number and type of reheats--Single versus double, gas versus steam.
3. Basic type of fuel element--Fission-product releasing versus fission-product retaining.
4. Type of helium circulator--Series-steam-turbine-driven circulator versus turbo-circulator.

Since all of the work on the fuel element, turbocirculator, core nuclear design, and concrete vessel was carried on as a part of Task II of the TARGET program and will be reported separately, this report on Task I will summarize the Task II effort only insofar as it affects the selection of a reference plant design.

4.1. CONCRETE VERSUS STEEL REACTOR VESSEL

In an effort to decrease capital cost and enhance the safety of reactor power plants, a study was initiated to search for construction materials that would offer alternatives to conventional approaches. The inherent advantages of the prestressing technique, in addition to articles presenting the merits of prestressed-concrete reactor pressure vessels in use in Europe,* indicated that an evaluation was desirable. Since European

* See for example J. Bellier and M. Tourasse, "Concrete Pressure Vessels," Civil Engineer, Vol. 13, February, 1959, pp. 71-75 and March, 1959, pp. 142-143; S. Gill and I. W. Hannah, "Prestressed

safety requirements and construction economics differ from those found in the United States, no direct method of comparison could be made on the basis of published reports. Comparisons were therefore made to investigate the claims of increased safety and lower cost.

In order to implement the comparison, it was decided to make preliminary PCRS (prestressed concrete reactor structure) and steel vessel designs at General Atomic, which were to include entire nuclear power plants, so that a comparison could be made not only of the costs associated with the primary containment but also of the other plant costs associated with the use of a PCRS design effort at General Atomic. The French firm of Grands Travaux de Marseille (this firm designed the prestressed concrete reactor structure for the third reactor unit (EDF-3) for Electricité de France) was retained to do a preliminary design study of a PCRS only. This arrangement allowed a direct comparison of a PCRS designed in accordance with prestressing criteria in the United States with a PCRS design by a firm that has completed a somewhat similar design which it is constructing for the EDF.

The steel pressure vessel design approach followed fairly conventional gas-cooled power reactor philosophies and is represented on Figs. 4.1 and 4.2. The reactor pressure vessel itself is spherical in shape with an internal diameter of approximately 47 ft. The steam generators are contained in cylindrical shells with vertical axes and are connected to the reactor pressure vessel by concentric ducts. The steam generators are located in a circular array outboard of the reactor pressure vessel. The reactor plant is secondarily contained, and all high-integrity pressure vessels are designed with requirements more stringent than stipulated in the ASME pressure vessel codes in order to accomplish acceptable safety standards. The pressure-temperature criteria were similar to those used for the PCRS and it was assumed that emergency cooling following primary system rupture could be achieved by a natural convection process.

For a more complete description of the design criteria utilized in the design of the PCRS, the reader is referred to Section 3.4. Because of the newness of this design approach, the design was carried to greater detail than was the design of the steel pressure vessels.

It was determined that the design of both the PCRS and the conventional steel pressure vessels was within the state of the art. There were several areas that would require attention to produce the most economical design for both the conventional steel reactor pressure vessel and the PCRS.

Concrete Pressure Vessels, " Nuclear Power, March 1962, pp. 48-50; and T. C. Waters and N. T. Barrett, "Prestressed Concrete Pressure Vessels for Nuclear Reactors," J. Brit. Nucl. Energy Soc., July, 1963, pp. 315-325.

Fig. 4.1--Plant arrangement (section A-A)

The design uncertainty for the steel pressure vessels relates to the number and spacing of the refueling nozzles that allow access to the core. It is believed that some experimental engineering would be necessary for a choice of the most economical design for the steel pressure vessel.

The design of the PCRS disclosed that existing stress analysis methods would create designs that were very conservative. In the matter of design and fabrication codes, it became apparent that only the conventional steel pressure vessel design has available a code of standard practice for the design and fabrication techniques. Comparison between the steel vessel and the PCRS were made in terms of relative safety and economics.

4. 1. 1. Safety

Although safety and cost are not completely separable, they have been separated in this report because safety is a paramount consideration. The studies thus far indicate that PCRS designs can readily be made to satisfy a wide variety of design philosophies related to safety. Naturally this produces cost variations in the PCRS cost as well as in total plant cost.

The safety philosophy that is most important in the design of the reactor system primary containment requires that no accidental or unpredictable event results in the release to the plant environs of quantities of fission products which could create a hazard to the general public. Present power reactors built in the United States conventionally use leaktight containment buildings around the primary containment under the assumption that primary system overpressurization or some other accidental event could result in an uncontrolled release from the primary system; thus, a secondary containment is provided to control undesirable fission-product releases from the primary system.

The present study was aimed at investigating the possible failure modes of a PCRS to determine whether or not the PCRS safety characteristics might require a conventional containment building. The philosophy was to provide a total plant, using a PCRS, with safety characteristics equal to those provided by the steel vessel concept.

The studies had limitations in scope because it was not practical to consider all geometries in detail. The possible failure modes related to safety are limited to a PCRS with a geometry similar to the reference

Fig. 4.2--Plant arrangement
(basement)

design. The geometry chosen for the reference design resulted from system and equipment requirements, and not because it gave a better PCRS design. In fact, the redundancies and discontinuities of the reference design geometry result in a PCRS that has a higher cost and that is more difficult to design and construct with the desired safety characteristics than would be encountered with other PCRS designs. Since this reference geometry is more difficult to cope with than others, the safety characteristics provided here would be even easier to achieve with less redundant geometries.

The design made and studied indicated that a steel inner liner would be desirable for several reasons: (1) Wide variation in published test data on the flow of gas through concrete and the necessity for extrapolating from test pressure to the reference plant design pressures did not give enough assurance that the leakage through unlined concrete would be or would remain as small as predicted. Where the whole internal surface of a PCRS is affected, the leakage uncertainty is too large to be desirable. (2) The steel liner thickness can be used as one means of varying the thickness requirements of the insulation inside the liner. (3) Some of the liner cost would be defrayed by its use in replacing formwork for the PCRS interior and penetrations. Only consideration (1) is related to safety.

Once the liner was conceptually established as a requirement, leakage from the PCRS was evaluated. It was evident that as long as the liner does not undergo large or small cracking as a result of excessive strain or strain repetition, the leakage from the PCRS would be no more than it would be for the steel vessel concept because, for crack-free steel, the gas leakage is so small as to be practically independent of thickness.

Since extensive cracking of the liner and concrete (akin to a brittle fracture rupture of a steel vessel) would play the largest part in influencing secondary containment requirements, this was evaluated by analysis and in the light of information on the hydraulic testing of concrete pipe and of a published report on the hydraulic testing of a model in the United Kingdom (see References, Section 4.1, above).

Since extensive cracking of the liner and concrete resulting in gross distortion of the vessel appears the most relevant to secondary containment philosophy, this will be discussed first. From the studies, it appears that such gross failures are incredible with the exception of singly contained penetrations. For this reason it appears that penetrations generally will need double seals, both of which can withstand the design pressure so that failure in one seal, or in the pipe enclosed in the penetration, will not

initiate a failure in the other. This requirement would be true only for a PCRS without more conventional secondary containment.

Although more penetration study is needed, present concepts for the more difficult steam penetrations dictate the use of double seals, both of which would be capable of withstanding design pressure. The outer seal would have a relief device, set well above design pressure, that would relieve to the atmosphere any steam pressure caused by a steam pipe rupture between the seals. In addition to this, the more probable problems of (1) reducing heat rejection to the concrete, (2) reducing the temperature of the metallic seals from steam pipe to ambient, (3) allowing thermal movements (with stops to prevent accidental thrusts from violating the seals), and (4) pipe system reaction problems must be taken into consideration.

The studies made to date indicate that gross vessel distortion (comparable to brittle fracture failure of a steel vessel) can be designed against with assurance. Reasons for this are mainly associated with attaining and maintaining tendon integrity and with the use of conventional reinforcing steel. Some of the reasons for this assurance are:

1. For system design pressures equal to or below the reference primary system pressure, any desired factor of safety may be attained with cost the only variable.
2. Unanticipated high-strength prestressing strand fracture in sufficient numbers to make failure imminent can be prevented by the usual installation prooftesting techniques and, if desired, periodic maintenance. Many methods are available to prevent changes in tendon strength owing to corrosion, and most of them would allow an inspection check on their preventive sufficiency. Because of the tendon elastic stress state and minimal stress variation, cyclic fatigue is no problem, and inspection could reveal this if any doubt remains. Radiation induced tendon damage is no problem, because adequate concrete is available for shielding. Prestressing caused by relaxation effects can be detected and corrected by inspection and retensioning should those losses create an unanticipated problem.
3. It may be credible to assume that some tendons fracture for an unknown cause. Since the tendons are individual units and not interconnected, the failure could not propagate from tendon to tendon (akin to brittle fracture) unless the vessel were grossly overpressurized. Hydraulic tests of a model in the United Kingdom, as well as prooftesting of prestressed concrete

pipes, suggests that the PCRS could act as a relieving device during its failure modes before complete collapse. As an indication of the reserve capabilities of multiple tendons, the reference PCRS has more than 8000 tendons. When stressed to 60% of ultimate tensile stress, approximately 4000 of them are necessary to maintain static equilibrium with the forces created by design gas pressure (assuming that the mild steel reinforcing and concrete do not resist tension). Although some undesirable concrete cracking might occur, it is believed that the vessel would not collapse at design pressure if several thousand tendons ruptured simultaneously. Since the attainment of design pressure would be extremely unusual, its attainment, with a simultaneous or undetected cumulative fracture of up to 4000 tendons, is not credible.

4. Since the concrete stresses can be made generally low and compressive when the PCRS is pressurized, much lower when it is unpressurized, a collapse owing to concrete failure alone is incredible when the PCRS is pressurized. This collapse is credible during prestressing owing to the proofloading of the concrete to a large fraction of its failing stress in compression. (Failure can be prevented by proper design.) A subsequent gross change in concrete strength can be guarded against by conservative design and by protection from overheating. At present, there are no known major radiation induced changes in concrete properties that are not attributable to overheating. Thus, it appears that with proper design the collapse of a PCRS is incredible while it is pressurized and that initial prestressing by tensioning tendons above design loads will check the concrete design at stresses approximately $1/3$ in excess of design stresses.
5. Poor concrete quality control has been considered as a possibility in gross PCRS collapse. Although not often used in ordinary construction, sufficient methods of attaining high concrete quality are known (and are commonly used by prudent owners or engineers) that lack of worker skills can be corrected by training. Although the prestressing operation itself is a concrete testing procedure, it is not a sufficient one; and any PCRS construction should use currently available concrete quality control techniques.

Since rapid depressurization by a major collapse appeared incredible, an evaluation was made of the possibility of depressurization as a result of cracking and by leakage other than instantaneous. The evaluation is difficult, because it is difficult to envision causes. The strain limitations imposed on the liner by the concrete (and the fact that the liner is generally

under compression) would tend to close rather than open cracks. Local buckling of the liner owing to compressive stress was examined, but the anchoring effect of penetrations, cooling coils, and additional anchors make this appear incredible. The credibility is also minimized by the relatively low compressive stress in the liner when the PCRS is pressurized, which would tend to reduce local bowing tendencies, and also by the fact that the gas pressure would positively push the liner against the concrete.

Since causes were difficult to envision, in the absence of gross overpressurization of the PCRS it was assumed that the liner cracked for no apparent reason. With concrete in intimate contact with the liner, there are two portions to a gas leak path. One is through the crack, the other portion is through the concrete. Since the flow in the liner crack is dependent on the pressure differential across the crack, and the concrete controls the pressure drop from the low-pressure side of the liner crack to a still lower pressure, it was evident that the pressure drop across the concrete must be known to find the flow condition in the liner crack.

In investigating gas flow in concrete, it was found that the little work that has been done in this area was for low pressure. No gas flow study for prestressed concrete was found. Some studies were found that related to nonreinforced concrete with air as the pressurizing medium.

All sources of information substantiated the correctness of the assumption that gas flow varied with concrete density and amount of cracking. The experimental test reports indicated wide variations in the flow rates.

Calculations made, using the largest experimental leak rates found, indicated that if 10 square feet of liner were removed, the leakage would be approximately 5 lb/hr for a 518 psig differential. This indicated that the flow would be controlled by the concrete. It should be emphasized that experimental results did indicate some variations, but that the leakage given was the highest calculated using the most pessimistic experimental results, which were even higher than test results on concrete with visible cracking. The compressive forces on the prestressed concrete should tend to reduce gas flow through concrete by pore-size reduction and by microfissure and crack reduction.

Since it is difficult to envision causes for liner cracking, let alone the loss of 10 square feet of liner, it appears that gas leakage through the reference PCRS walls should be small enough to be manageable by means other than a high-pressure resisting containment. As a scale of reference to the 5 lb/hr helium leakage calculated above, the reference PCRS at operating pressure contains 9400 lbs of helium.

Gross overpressure in the PCRS would be prevented by positive means, just as it is in the case of steel pressure vessels by

provision of multiple safety relief valves.

Because the large-scale leakage of a PCRS does not appear to be credible, differing secondary containment philosophies could be used on plants using a PCRS compared with plants using steel pressure vessels.

Even though a large-scale leak does not appear credible with the PCRS, it cannot be guaranteed that a small leak will not occur. As is shown above, however, a leak will be sufficiently small that the rate of pressure fall in the pressure vessel would not cause serious difficulties in cooling the core in an emergency, and that the leak will be such that it would be collected in the leak collector and pumped back into the primary system or storage system by means of the helium transfer compressors.

Before this philosophy can be finally accepted, it will be necessary to carry out extensive tests with concrete models to prove the various points to the satisfaction of those responsible, and it is intended that this will be done. The statement that a gross failure cannot occur is predicated upon the assumption that a serious increase in system pressure cannot occur. The concrete vessel, however, has a high factor of safety inherent in its design, and it is not difficult to ensure that the system pressure will never exceed a maximum emergency figure.

With a steel vessel, the containment must hold the complete primary system contents in the event of a rupture, thus resulting in high cost. This containment cost is further increased by the higher helium inventory in the steel vessel system. In addition, their higher inventory increases the hazard due to stored energy.

In order to reflect the extreme in conservatism, and thus assess the effect of that extreme on the total plant, the assumption was made in one case studied that it would be impossible to adequately demonstrate the ability of a PCRS to control leakage and not fail catastrophically. On the basis of this assumption, a containment building capable of fully containing the products of any credible incident was included in the case.

Since it does not seem plausible that the safety capabilities of a PCRS cannot be adequately established, all feasible ways of establishing full containment of an uncontrolled primary system release were not considered and only one possible means was briefly examined.

The containment concept briefly examined is shown, in concept form, on Fig. 4.3. It is somewhat analogous to the concept established for the steel reactor vessel cases but assumes the use of a steel-lined reinforced concrete shell instead of a steel shell. The larger containment free volume in conjunction with the lesser helium inventory achievable with a PCRS results in a lower incident pressure than was considered for the steel vessel cases.

In concept, the containment shown on Fig. 4.3 differs from the reference plant in that the "leak collector" and the conventional building are dispensed with. In their place is substituted a pressure-resisting shell in which the concrete and reinforcing steel combine the functions of shielding under incident conditions with the materials for resisting external plus internal loads and in which the steel liner membrane ensures compliance with leak standards. Since the containment shell may safely retain the products released by various credible incidents, safety as related to the uncontrolled release of fission products is attained in much the same manner as visualized in the steel reactor vessel cases.

As is always the case, the increase in anticipated safety results in increased costs. As an illustration of the increased cost, the decrease in savings which would result from provision of a full secondary containment in the reference case amounts to approximately \$3/kw(e), which is the cost of the secondary containment decreased by the cost of the leak collector and the conventional building it replaces. (Only the conventional building exterior walls are replaced by the shell.) This \$3/kw(e) is less than the containment cost for the steel vessel cases as cited in Section 4.1.5 and still leaves an estimated net savings in plant total capital cost of approximately \$14.00/kw(e). Although the \$3/kw(e) may appear to be a relatively small sum, it is still a large premium to pay for an extra redundancy in safety, and the inherent safety of a PCRS should be thoroughly examined before a set of assumptions leading to extensive secondary containment features is established.

4.1.2. Economics

As a result of a comparison undertaken in this study, it appears that a savings of approximately \$17/kw of generating capacity can be achieved by utilization of the PCRS design approach. This is equivalent to approximately 13% to 14% of the total plant cost. The magnitude of this estimated cost saving can be appreciated when it is noted that the total amount of saving is approximately equal to the purchase price of the turbine-generator for this plant. As might be expected, all of the estimated cost savings are not found in the difference in cost between the steel pressure vessel and the PCRS alone. In fact, the magnitude of the estimated total saving is somewhat surprising when considered in the light of the total cost of a PCRS.

Fig. 4.3--Representation of alternative containment

In order to illustrate the areas in which the estimated cost savings are found, the various estimated cost differences were extracted from their codes of accounts and are briefly explained in subsequent subsections.

Tables 4.1 and 4.2 show that the cost savings are not so much the result of using a PCRS alone, but are the result of possible changes in total plant design that utilize the capabilities of a PCRS. Any steam generator or reactor design that does not efficiently utilize the interior volume would not only raise PCRS costs but would raise total plant costs. It is axiomatic that different total plant designs (and less well-planned, integrated, and designed systems and layouts) could well show negative cost savings.

4.1.3. System 10 -- Reactor Systems

Table 4.1 indicates that this portion of the code of accounts shows a minus \$1.68/kw cost savings. This is not strictly correct, owing to the method used in the code of accounts. Item III of Table 4.2 indicates that the PCRS has 14,160 yd³ of concrete. For the reference plant, the code of accounts charges this concrete to System 10 as a part of the PCRS. Case 2, which is a comparable plant using the steel vessel concept, charges similar concrete to System 70 (Structures and Improvements) as a part of reactor building cost. (In the reactor building account this concrete is partly structure and partly shielding.) For the comparison of total plant costs, comparisons of the steel primary containment costs versus a PCRS cost is academic. There are so many factors that affect total plant cost that a judgment as to whether the cost change resulted from using a PCRS is somewhat subjective. However, were the concrete cost for the PCRS removed from System 10 and added to System 70, and the steam generator shell costs removed from System 20 and added to System 10 for a comparable steel vessel concept, one might show that a PCRS was cheaper than a steel vessel when compared on an equivalent function basis rather than by an arbitrary code of accounts.

4.1.4. System 20 -- Heat Transfer Systems

The major items in System 20 which result in large cost savings are the steam generator shells and helium storage tanks. For the reference case, the steam generator shells are the PCRS, so no cost item appears. In addition to the shells, other cost reductions appear in the elimination of the concentric ducts, in support costs, and in other minor items.

Table 4. 1

BREAKDOWN OF DIRECT AND INDIRECT COST SAVINGS

<u>DIRECT COST SAVINGS</u>	<u>\$/kw</u>
System 10. Reactor Systems	
11. Reactor vessel and internals	-1.72
16. Auxiliary equipment	0.04
Subtotal of cost savings:	-1.68
System 20. Heat Transfer Systems	
21. Primary coolant (including helium circulators)	5.79
24. Helium storage	0.34
Subtotal of cost savings:	6.13
System 40. Plant Water Systems	
46. Shield cooling water system	0.05
Distributed*	0.03
Subtotal of cost savings:	0.08
System 70. Structures and Improvements	
71. Reactor building	
Reactor building(only)	1.44
Secondary containment	4.32
Interspace purification	0.29
Interspace cooling	0.40
72. Reactor service building	0.10
73. Turbine building	0.18
Subtotal of cost savings:	6.73
Algebraic sum of direct cost savings (12.94 - 1.78)	\$11.26/kw
<u>INDIRECT COST SAVINGS</u>	
General and Administrative	0.55
Miscellaneous Construction Costs	0.12
Architectural and Engineering Services	1.14
Nuclear Engineering	0.12
Contingency	1.31
Interest	2.96
Total Indirect Cost Savings:	6.20
Sum of Direct + Indirect Cost Savings = \$11.26 + \$6.20 = \$17.46kw	

* These refer to a sum of many cost savings as explained in Section 4. 1. 5.

NOTE: The above figures use the AEC ground rules for indirect costs, and all direct costs include Payroll Burden Expense (taxes, insurance, and fringe benefits).

Table 4.2

COMPARISON OF PCRS AND STEEL VESSEL CONCEPT
MATERIAL QUANTITIES

Item	PCRS (Case 3)	Steel Vessel (Case 2)
I. Helium in primary system, lb	9400	14100
II. Steel, primary and secondary containment, tons		
Pressure resisting steel for primary system containment (ducts + pressure shell and liner + prestressing strand)	2630 [*]	2500 [†]
Steel for secondary containment	360	4200
Miscellaneous steels:		
Reactor support (including core barrels, thermal shields, etc., as required)	250	300
Internal ducting (reactor and steam generator)	210	230
Support steel (steam generator supports, tendon encasements, and bearing plates, etc.)	1530	80
Stainless steel insulation	70	40
Helium storage tanks	470	890
Additional dollies		30
Structural steel	<u>680</u>	<u>1120</u>
Total steel tonnages	6200	9390
III. Concrete in place (including reinforcing steel), yd ³ :		
PCRS	14160	0
Structure and shielding	<u>11000</u>	<u>36300</u>
Total concrete	25160	36300

^{*}Includes 2200 tons of high-strength prestressing strand.

[†]Includes 1700 tons for steam generator shells.

Table 4.2 shows that the primary system helium inventory is 9400 lb for the reference case and 14,100 lb for the comparable Case 2 steel vessel concept. Since the refueling is done at essentially 0 psig, the helium is handled and stored elsewhere during refueling. Thus, the reference case requires less storage facilities. Other savings could result in the gas handling systems if equivalent refueling depressurization or repressurization times are specified.

4.1.5. System 40 -- Plant Water Systems

Table 4.1 shows some of the minor savings that appear. The two major reasons for the cost reduction are (1) that part of its equivalent is replaced by the cooling coil circuits in the System 10 PCRS costs and (2) that the total heat removal is less. The item listed as "distributed" on Table 4.1 is the sum of many small cost reductions..

4.1.6. System 70 -- Structures and Improvements

By far the greatest cost savings in this portion of the code of accounts is the secondary containment. Table 4.1 shows this to be \$4.32/kw. Since the omission of a more conventional secondary containment may be debatable, this savings may be also. To evaluate the effect of a more conventional secondary containment, see Section 4.1.1. There it is shown that direct cost increase for the more conventional secondary containment would be \$3/kw, thus still leaving a \$1.32/kw net savings. The \$3/kw was estimated as a separate entity and uses a large steel-lined concrete shell.

In order to evaluate the comparison, it should be noted that the helium inventory for the reference plant is 9400 lb and for Case 2 it is 14,100 lb (see Table 4.2). Since the primary system helium inventory plays a large part in the size and cost of the secondary containment, a comparison of the two should give secondary containment costs which are somewhat related to helium inventories (providing the secondary containment volumes displaced by shielding, etc., are comparable). Thus, the reference plant, with conventional secondary containment, should have a lesser cost than a plant with a higher primary system helium inventory. This would tend to indicate that the \$1.32/kw net savings for the reference plant secondary containment (assuming a more conventional pressure resisting type) has validity as a savings, with only the amount of savings in question.

On Table 4.1, System 70 shows a \$1.44/kw savings for the reactor building. Part of the reason for this (see Table 4.2) is that the amount of building structural steel and concrete is less because the building is smaller. A second reason is that the 14,160 yd³ of concrete entered in System 10 as a part of the PCRS cost, does not appear in the cost of the reactor building for the reference concept. Thus, a portion of the concrete, with functional similarity to Case 2 reactor building concrete, is contained in System 10 accounts, and is costed with the PCRS for the reference case. This again emphasizes the point that the multipurpose use of a PCRS makes difficult an assessment of the difference between it and the pressure resisting equivalent steel vessels.

The interspace cooling and purification systems show small savings. Among the reasons for this is the fact that the total surface area of the PCRS from which heat is rejected is less than that of the Case 2 steel reactor vessel plus concentric ducting plus steam generator shell surface area. In addition to this, the PCRS interior insulation, the liner cooling coils, and the insulation provided by the thick concrete walls not only are paid for in System 10 rather than in System 70, but reduce heat loss from the PCRS outer concrete surface to a minimum.

The cost savings for the reactor service building and turbine building are small and are the result of many small items.

4.1.7. Indirect Cost Savings

Most of the indirect cost savings are a function of AEC ground rules. These ground rules require some indirect costs to appear as predetermined percentages of the previously described direct costs (see Table 4.2). Since the direct costs for the reference plant are estimated as less than the comparable Case 2, the indirect charges are naturally less when the percentages are constant.

The interest charges are affected by a factor which other indirect costs do not reflect--the construction period. It is estimated that the reference plant could be constructed in a shorter time period owing to less total construction (indicated by less direct construction cost) as well as some concurrent construction that is not possible with the Case 2 comparison. Thus, smaller direct construction costs, smaller indirect construction costs, plus a shorter interest period reflect a lower interest cost, even though the interest rates would be the same. If the interest periods were identical for the reference case and its Case 2 comparison, the total interest charges would still be lower because of the lower direct plus indirect costs and a constant interest rate.

4.1.8. Cost Savings Evaluation

Cost estimates do not always reflect the constructed cost, especially when the estimates are not made from completely detailed designs and when the construction contains new or unusual concepts. The PCRS was designed with sufficient detail that all major cost items could be estimated. The unit quantity estimate was compared with the estimate received with the preliminary design by Grande Travaux de Marseille. Correlativity was high in comparable items, and comparable items were the major cost items.

It could be argued that the PCRS itself will cost more than estimated owing to its newness in the United States. This is provided for by several contingency items. Among the largest items are those in the indirect costs. Other contingencies result from the fact that certain cost reductions that appeared after drawings were completed for estimate were not entered into the cost summary.

In addition to the contingency items, another factor that tends to substantiate the PCRS estimate is the prestressing tendons. The estimate contains a quotation from a large commercial supplier of prestressing systems. The tendons are in lengths shorter than used in many other structures, so they are not unusual. The tensioning operations are standard and are accomplished by partly mechanical means that are fairly well standardized. The tendons are used in identical multiple units, so mass production techniques should reduce costs. At least two prestressing systems were estimated (with similar costs), so the tendons can be bid competitively. These facts, coupled with the fact that over 50% of the PCRS cost is for the prestressing tendons, should tend to keep the construction cost within the estimate. The more variable costs would be for units whose total cost is less than 50% of the estimated cost of the PCRS.

The concrete is a small part of the PCRS cost, and its actual cost could vary widely with little effect on the total.

Where the systems were directly comparable, the reference plant used the same costs as the equivalent steel vessel plant. Where different, they were estimated in the same manner. Thus, the cost difference will probably remain regardless of small unit cost variations.

Designs involving the addition or subtraction of all or portions of other systems can cause differences in the estimated cost. For that reason only the cost differences were evaluated, under the assumption

that additions and subtractions where costs did not differ would have little effect on the anticipated cost savings. An example of this is the addition of a more conventional secondary containment. (This is a large part of the total direct, as well as indirect, cost savings.) This addition still does not reduce the estimated cost savings to \$0/kw. Since the change in the largest cost savings will not reduce the savings to nothing, it appears that no probable combination of changes will reduce the cost savings to nothing.

True cost differences could only be ascertained by constructing several of each concept compared. For future prediction from present cost estimates, it appears that the \$17/kw savings has validity and, for the compared designs, cost savings of some amount are almost certain.

4.2. COMPARISON OF TURBOCIRCULATOR WITH SERIES-STEAM-TURBINE-DRIVEN CIRCULATOR.

As a part of Task II of the TARGET program, a complete aerodynamics and mechanical preliminary design for a turbocirculator was carried out. The details of this study will be reported in a separate report.

Figure 4.4 shows the general arrangement of the turbocirculator machine and Figs. 4.5 through 4.7 show the location of the turbocirculator in the reactor plant. Six circulators are used in the 1000-Mw(e) TARGET plant. The vertically oriented machines are approximately 22 ft in over-all length and have a compressor tip diameter of 4 ft. For the pumping power used as a basis for design, the compressor has three stages and the turbine one stage. These machines have oil-lubricated bearings with an integral oil system.

The design of the series-steam-turbine circulator and its location in the reactor plant is described in Section 3.5.

Some insight into the relative advantages of the two types of circulators can be had from the following theoretical thermodynamic comparison.

In an HTGR-type power plant such as TARGET where the steam conditions are not limited by the reactor gas temperatures, the turbocirculator has the potential of providing the required pumping power thermodynamically without penalty to the net plant efficiency. For the same net electrical send out, externally driven circulators such as the series steam turbine require additional heat in the reactor and steam generator and additional steam flow in the conventional portion of the plant. The additional reactor heat is

$$\frac{\Delta q_{in}}{q_{in}} = \frac{\lambda}{\eta_c} \left(\frac{1 - \delta e_T}{\delta e_T} \right) ,$$

where q_{in} = reactor thermal output,

λ = flow loss in the primary gas circuit expressed as the fraction of thermal heat produced in the reactor,

δ = a coefficient which accounts for the difference in producing work in the steam plant and in the circulator drive,

Fig. 4.4--Turbocirculator installation

Fig. 4.5--Reactor assembly layout
(plan view)

Fig. 4.6--Reactor assembly layout
(section A-A)

Fig. 4.7--Reactor assembly layout (section B-B)

= 0.934 for the series-steam-turbine drive,

e_T = the thermal efficiency of the plant, and

η_c = efficiency of the helium compressor in the externally driven compressor.

The additional heat in the steam generator and the additional steam flow in the steam plant is equal to this additional reactor heat plus the thermal equivalent of the circulator work:

$$\frac{\Delta W_s}{W_s} = \frac{\Delta q_{in}}{q_{in}} + \frac{\lambda}{\eta_c} ,$$

where W_s = the steam flow.

The relationship between thermal efficiency for externally driven circulators and the turbocirculator is given by

$$\epsilon_T = \frac{e_T}{1 + \frac{\lambda}{\eta_c \delta^{e_T}} - \frac{\lambda}{\eta_c}} .$$

The increase in reactor heat required, the steam generator heat transferred, and the steam flow required in the case of the series-steam-turbine circulator are shown in Fig. 4.8 as a function of pressure drop in the primary circuit for the identical thermodynamic state points. In an actual plant design these state points will vary as required by design optimization and in many instances, the costs connected with increased capacity tend to be step functions rather than monotonically increasing ones. Nevertheless, at the circuit pressure loss of $\Delta p = 9$ psi, approximately that of TARGET, the additional reactor heat required is 2.17%, which is also the loss of thermal efficiency due to the externally driven circulators. The increase in steam generator heat transferred and steam flow in the conventional plant would be 3.74%. If the pressure drop were doubled, $\Delta p = 18$ psi, the reactor heat would have to be increased 4.27%, and the steam generator heat and conventional plant steam flow increase would be 7.32%. However, a higher pressure drop would have to show advantages to the plant to be considered. Studies of possible advantages for plants with a higher pressure drop have not been performed.

In addition to the above thermodynamic comparison, a comparison between the series-steam-turbine helium circulator and the turbo-

Fig. 4.8--Incremental reactor heat and steam flow increases as a function of primary system pressure drop

circulator was made by estimating the cost of a reactor power plant using each type of circulator. These estimates were made for the identical reactor power plant except for the differences that result from each type of circulator. The same reactor thermal output was assumed which resulted in a higher net electrical send out of 1.5% and thermal efficiency of 46.5% versus 45.8% for the case of the turbocirculator and an increase of steam flow of 3.75% for the series-steam-turbine plant. The major items of cost difference in the plant result from the increased heat transfer surface required in the steam generator and some savings on the ducting in the case of the series steam turbine. Since at this time the cost of producing turbocirculators can only be estimated on the basis of the General Atomic preliminary design, the results of this study are shown in Fig. 4.9, where the unit capital cost is plotted as a function of additional turbocirculator cost over that estimated for the series steam turbine. Figure 4.9 shows that the plant capital costs of both systems are equal when each turbocirculator costs \$313,500 more than each series steam turbine, or \$1,881,000 for the six units in the plant. For an equal cost of both machines, which is unlikely since the turbocirculator is more complex and therefore more expensive, the total savings in capital costs would be \$3,200,000. These figures do not include the fuel savings which result from the improved thermal efficiency.

The comparison of the plant economics for both circulators is summarized in Cases 3 and 4 on Table 4.18 for an assumed turbocirculator cost of \$300,000 each in excess of the cost of the series-steam-turbine-driven circulator.

In addition to its effect on capital cost, the turbocirculator also affects fuel cost by virtue of its effect on plant efficiency. The magnitude of the effect depends on the fuel cost. For example, if the fuel cycle cost with a series-steam-turbine-driven circulator were 1.0 mill/kw-hr, then the use of a turbocirculator would reduce the fuel cost approximately 0.015 mill/kw-hr. The capitalized value of the saving is approximately \$0.75/kw.

Based on these studies, it does not appear that the potential savings that would result from using a turbocirculator justify its development cost. The reference plant design is therefore based on the use of a series-steam-turbine-driven circulator.

Fig. 4. 9--Plant capital cost: turbocirculator versus series steam turbine
(variable turbocirculator cost increment over series steam turbine)

4. 3. CHOICE OF FUEL ELEMENT

4. 3. 1. Fuel Element Design

4. 3. 1. 1. Factors Influencing the Design. A major goal in the TARGET fuel element design effort has been to utilize wherever possible the design information developed under the Peach Bottom reactor program. The Peach Bottom fuel element is completely graphite moderated and contains coated fuel particles that are relatively retentive for fission products. Those fission products that do escape from the particles are swept by a purge gas stream to fission product traps located outside of the reactor core. This design concept (a purged, graphite clad fuel element) has been the basis for all but one of the designs studied under the TARGET program. The exception was a bare unpurged fuel element which appeared to be of interest because of its low operating temperature and simple construction.

Besides the goal of advancing the design technology begun under the Peach Bottom program, a second major goal of the TARGET program has been to achieve a breeding or near-breeding fuel cycle. In order to gain the high conversion ratio necessary for a near-breeding reactor, three basic design features not used in Peach Bottom fuel elements have been actively pursued under TARGET; These three features are (1) to replace a significant fraction of the graphite moderator inside the element with beryllium oxide, (2) to utilize fuel which releases large fractions of the volatile fission products from the active core, and (3) to accommodate fuel in a form amenable to efficient recovery and refabrication (i. e., recycle) after irradiation.

The use of beryllium oxide as a fraction of the moderator material appears desirable because of the improved neutron moderation characteristics of beryllium oxide over graphite. While the use of this material may not reduce fuel cycle costs significantly, its use does notably improve the conversion ratio.

Fuel that readily releases fission products has long been considered advantageous to the conversion ratio of a reactor. The removal of high cross-section fission products such as Xe^{135} and Kr^{83} provides the improvement in neutron economy. Although the possible gains are significant, the technical problems associated with developing a highly releasing fuel are great. Design work has proceeded on alternative concepts for highly releasing fuel, even though the reference design has continued to utilize fission-product retaining fuels.

The use of recycled fuel has continually affected the fuel element designs. Provisions have had to be made to partially recycle fuel in

order to limit the buildup of U^{236} and Np^{237} poisons. The need for partial recycling and the methods for accomplishing it have received considerable attention.

For any reactor operating on the thorium-uranium fuel cycle in which uranium recycle is considered, some fraction of the equilibrium cycle power is generated by U^{235} . If the reactor has a conversion ratio of unity, or if U^{233} feed material is available, then the fractional fuel absorptions in U^{235} will take on its minimal value of $\alpha/(1 + 2\alpha)$ (where α is the capture-to-fission ratio in U^{233}). The fractional fuel absorptions in U^{235} is a function only of the reactor spectrum and conversion ratio when U^{235} feed is considered (see Fig. 4.10) for the TARGET reactor. Now for each absorption in U^{235} , roughly 0.2 atoms of U^{236} are formed, which in turn absorb neutrons to form Np^{237} . If the residence time is very large, the Np^{237} will equilibrate, and then for each absorption in U^{235} roughly 0.4 neutrons will be parasitically lost. The neutron losses to Np^{237} can largely be eliminated by frequent chemical processing; however, the processing losses and costs would become exorbitant and would have no effect on the neutron losses in U^{236} . If the uranium could be separated in such a way as to limit the number of times it is cycled through the reactor, neither the U^{236} nor the Np^{237} would equilibrate and the neutron losses would be reduced significantly. For this reason, we have considered fuel segregation.

The segregation of the three fuel components (thorium, uranium produced during the previous cycle, and fully enriched uranium feed) among two separable fuel particles is shown schematically in Fig. 4.11, together with complete uranium recycle; they are compared qualitatively in Table 4.3.

Table 4.3

MODES OF FUEL SEGREGATION

Type	Particles		Particle Fabrication Cost	Burnup on Some Particles (%)	Fraction of Particles or Compacts with Higher than Average Heat Flux (%)
	A	B			
I	Th	Bred U Fed U	Low	90	10
II	Th Bred U	Fed U	High	90	1
III	Th Fed U	Bred U	Low	90	9

Fig. 4.10--Fractional fuel absorptions in U^{235} as a function of conversion ratio

Fig. 4.11--Schematic description of fuel segregation

The advantage of Type I fuel segregation is that all the thorium particles are made in the absence of uranium (including U^{232}). Therefore, there is no criticality or γ -ray hazard; however, all of the heat is generated in the B particle initially, and the heat-transfer problems could be formidable. The B particles could not be diluted with graphite to normal uranium density because of the resulting increase in reactor size. The advantage of Type II recycle is associated with the small volume associated with the B particles and the large fractional power always generated in the A particles. The heat transfer problems could be alleviated by diluting the B particles with graphite, and this would require only a small increase in reactor size. However, the costs are higher, since the thorium particles must be made under the restrictions of criticality and the γ hazard associated with the U^{232} in the bred uranium. Type III recycle has many of the disadvantages of both Types I and II and is consequently of little interest.

A comparison of the neutron balances for full recycle and the three types of fuel segregation are shown in Table 4.4 for a four-year cycle time with $Be/Th = 28$. Type I fuel segregation results in the largest reduction of losses of U^{236} and Np^{237} and consequently in the highest conversion ratio. The conversion ratio advantage of Type I recycle over Type II is only 1%, and the heat transfer problems associated with segregated fuel might dictate the use of Type II recycle.

Table 4.4

COMPARISON OF NEUTRON BALANCES WITH FUEL SEGREGATION*

	Full Recycle	Type I	Type II	Type III
η	2.07	2.12	2.10	2.11
$\epsilon\eta$	2.15	2.20	2.18	2.19
Leakage	0.02	0.02	0.02	0.02
Moderator (including Li^6)	0.06	0.08	0.07	0.08
$U^{236} + Np^{237}$	0.09	0.01	0.04	0.02
$Pa^{233} \times 2$	0.02	0.02	0.02	0.02
$Xe^{135} + Sm^{149}$	0.04	0.05	0.04	0.05
Fission product points (others)	0.06	0.08	0.07	0.07
Control	0.04	0.04	0.03	0.03
Total losses	0.33	0.30	0.29	0.29
$\epsilon\eta - 1 - L$	0.82	0.90	0.89	0.90

*These neutron balances correspond to $Be/Th = 28$, cycle time = 4 years, fission-product retention, GA cross sections.

The actual fuel segregation could be accomplished either by using separate fuel compacts or by using separable fuel particles. If separate compacts are used, the separation of the irradiated fuel would be relatively simple, but thermal difficulties might be encountered in the compacts that do not contain thorium. If segregation is accomplished by using separable fuel particles, then the thermal difficulties are minimized. The actual particle separation could be accomplished in a number of ways. Three possible methods are:

1. Using particles of different size and separating by screening,
2. Using coatings which can be differentiated chemically, and
3. Using the difference in the chemical reaction rates of the ThC_2 - UC_2 solid solution and UC_2 alone.

In all of these methods one would expect a certain fraction of cross mixing between the two particles. This would result in only a small decrease in the conversion ratio (less than 0.01 if the mixing is kept below 30%), and thus the separation process can be relatively inefficient.

Nine different fuel element designs have been studied under the TARGET program. All are capable of utilizing recycled fuel. For discussion purposes, the different designs have been grouped by types according to fission-product retention and the moderator material. The four resulting types are (1) fission-product retaining with all-graphite moderator, (2) fission-product retaining with beryllium oxide spine, (3) fission-product releasing with all-graphite moderator, and (4) fission-product releasing fuel with beryllium oxide spine. These four types are described in the following four subsections.

4.3.1.2. Fission-Product Retaining Fuel with All-graphite Moderator.

This fuel element type embodies a natural extension of current Peach Bottom technology. Present-day knowledge would allow fabrication of a core to meet requirements of this technology. An enlarged Peach Bottom fuel element and three more advanced fuel elements have been studied under this type. Cross-section views of these four designs are shown in Figs. 4.12 through 4.15.

The enlarged Peach Bottom (see Fig. 4.12), the bare unpurged (see Fig. 4.10), and the central cooled annular (see Fig. 4.15) all utilize fuel bodies made by compacting fuel particles within a graphite matrix. For these designs, the recycled uranium from makeup U^{235} would be separated by fabricating separate fuel bodies, with one type containing the recycled fuel and the other containing the makeup U^{235} .

Fig. 4. 12 -- Enlarged Peach Bottom type element

Fig. 4. 13--Bare, unpurged fuel element with BeO

Fig. 4.14--Packed-bed fuel element

Fig. 4. 15 --Central cooled annular fuel element; can be self-purged

The packed bed fuel element (Fig. 4. 14) utilizes beds of fuel particles packed within an extruded graphite body of a slotted shape. With this design the fuel for recycle can be segregated from makeup U^{235} by utilizing two different size ranges of particles. Particles initially containing only thorium or thorium plus U^{233} could be of 350 to 500 microns in diameter, for example, while the particles containing U^{235} could be 150 to 250 microns in diameter. This type of fuel segregation has a disadvantage in that a fraction of the fuel particles (those containing only uranium) experience high power densities during portions of the fuel lifetime. This could cause greater particle or coating failure during operation.

4. 3. 1. 3. Retaining Fuel with Beryllium Oxide Spine. The most logical step in designing an element in this category is to replace the graphite spine of an enlarged Peach Bottom element (Fig. 4. 12) with a beryllium oxide spine. More advanced designs in this category are the packed bed annular (Fig. 4. 16) and the multihole packed bed (Fig. 4. 17). The latter design has been called the "Telephone Dial" shape. This is the design which was eventually established as the reference design and therefore, is described in more detail in Section 3. 1.

The means for segregating recycled fuel in the two packed-bed designs presented in Figs. 4. 16 and 4. 17 is by providing two different particle size ranges as was proposed in Section 4. 3. 1. 2. for the packed-bed fuel element.

4. 3. 1. 4. Releasing Fuel with All-graphite Moderator. Designs proposed for this type of core are the releasing fuel element (Fig. 4. 18), the releasing fuel element with body in sleeve (Fig. 4. 19), and the annular releasing fuel element (Fig. 4. 20). In both of the five-hole geometries the graphite bodies are extruded to shape. In the design shown in Fig. 4. 19, the graphite sleeve is added as an alternative way of obtaining an outside cladding which is a good retainer of fission products while the five-hole body remains highly permeable to fission products moving toward the center purge hole. In each of these designs the purge gas enters the fuel elements radially and flows downward through the center holes.

The properties of a fuel body which readily releases fission products are not well understood at present. Considerable materials development work on this concept must be performed before it can be seriously considered for a reference design of a reactor system.

The critical development item in this concept is the fission-product-releasing fuel matrix. The most favorable fuel tested for release was

Fig. 4.16--Packed-bed annular fuel element with BeO

Fig. 4.17--Multihole-body fuel element with BeO
(reference design)

Fig. 4.18--Releasing fuel element

Fig. 4. 19--Releasing fuel element with body in sleeve

Fig. 4.20--Annular releasing fuel element with graphite spine

made by dispersing fine ThO_2 and UO_2 particles (~ 1 micron diameter) within microfine graphite (~ 0.5 micron particles). Initial release rates have been high with this material; however, sintering at 1500°C has shown a reduction in release rates. Considerably more development work is necessary to determine the feasibility of a fuel element that releases a sufficient quantity of the fission products to improve substantially the neutron economy. However, if such a fuel element can be built, its use probably will provide decreases in fuel costs as well as an increase in conversion ratio.

With each of the designs for releasing type fuel, the means of segregation for partial recycling of fuel is to utilize separate axial zones in a given fuel column for the recycled uranium and for the makeup U^{235} .

A significant design feature of the fission-product-releasing fuel concept is the use of relatively large fission-product traps directly below the fuel elements within the reactor vessel. Each fuel element rests upon a trap which consists of an 8-ft long metal can, 4.5 in. in diameter, and which contains charcoal. These traps hold up fission products for several seconds and allow for the decay of short-lived fission products. Approximately 40 Mw of heat are recovered from these traps by the incoming helium coolant.

4.3.1.5. Releasing Fuel with Beryllium Oxide Moderator. Two fuel element designs have been studied for this category. These are the multihole body (Fig. 4.17) with releasing fuel replacing the coated particles and the annular releasing with BeO spine (Fig. 4.21). In both of these designs the purge gas enters the fuel element radially through the slightly permeable graphite. The method of fuel segregation is by utilizing separate axial zones in a given fuel column for the recycled and the makeup uranium.

4.3.1.6. Utilization of Graphite Extrusions. In most of the designs other than the enlarged Peach Bottom type, the development of extruded graphite bodies or fuel matrices is required. Considerable extrusion development work has been performed at General Atomic as part of the TARGET program. To date, 19-ft long sleeves of 3-1/2 in. outside diameter have been extruded to the green stock stage. Long sections of 3-1/2 in. multihole geometry have been extruded. Because of equipment limitations, the extrusions have been cropped and baked in 4-ft sections. "The possibility of extruding precision shapes of multihole design for the fuel-element bodies appears good. The use of extruded graphite appears to be highly desirable for meeting the requirement of utilizing materials in forms requiring simplified fabrication methods.

Fig. 4.21--Annular releasing fuel element with BeO spine

4.3.1.7. The Analysis of Fuel-Element Concepts. Thermal analysis work has been performed on the advanced fuel-element concepts considered for TARGET. The general aim of the thermal design has been to limit the long-term fuel temperatures to 1500°C in the case of coated-particle fuel, and to limit the boundary temperature between oxide fuel and graphite to 1500°C in the case of oxide fuel. The summary of the maximum fuel temperatures for the different fuel element designs is presented in Table 4.5. Also shown in this table are the core coolant pressure loss values for each design. In all cases the reactor was assumed to be producing 2270 Mw(t) of power. The fuel element pitch, coolant conditions, core geometry, and neutron flux profiles were comparable in each case. The data in Table 4.5 are shown for comparison of the thermal performance of competing fuel element designs. In those cases where the fuel temperature exceeds the design limit of 1500°C , reductions can be made by closing the fuel element pitch, by applying local surface roughening, or by utilizing axial zoning of fuel loadings. The corrective measures would result in higher core pressure drops and/or higher fabrication costs.

A number of basic assumptions were used in making the thermal calculations. The radial peak-to-average power was 1.30, and the axial power profile was a modified cosine curve with secondary power peaks at either end and a central-peak-to-average power of 1.32. The thermal conductivity for graphite was based on a curve which decreased with temperature. A representative value of the graphite thermal conductivity of packed coated particles increases with temperature over the range of interest in packed-bed fuel elements. A representative value for the packed bed is 1.5 Btu/hr-ft $^{\circ}\text{F}$ at 2600°F . For these calculations, the core inlet coolant temperature was assumed to be 750°F with an average core coolant temperature rise of 750°F .

Analytical calculations were performed to determine the main coolant activities of fission-product gases released from the different fuel-element designs. Relative values of the fission-product levels in the main coolant are shown in Table 4.6. These data are approximate and should be used only for comparing fuel-element performance. In calculating these data the purge-flow rate per fuel element was assumed to be 0.7 lb/hr. The fission-gas release from coated fuel particles was based on release fractions from the GAIL III-B test element. Of the total purge gas flowing through the fuel elements, 30% was assumed to pass through a liquid-nitrogen cooled trap. All of the purge gas was assumed to pass through delay beds which would hold up all but the noble-gas fission products (the latter being delayed). For the purged designs, considerably lower gaseous activity can be achieved by use of a trapping system in which 100% rather than 30% of the purge flow is completely cleaned up. However, based on presently developed adsorption-type trapping systems, this reduction in gaseous activity would be accompanied by a large increase in plant capital cost.

Table 4.5

THERMAL PERFORMANCE COMPARISON

Fuel Element Design	Figure No.	Maximum Fuel Temp. ($^{\circ}$ C)		Core Pressure Drop (psi)	
		4.5 in. Diam.	4.6 in. Diam.	4.5 in. Diam.	4.6 in. Diam.
Enlarged Peach Bottom	4-7	1535	1460	3.5	5.5
Nonpurged	4-8	1350	1275	3.5	5.5
Packed-bed, slotted body	4-9	1600	1522	3.5	5.5
Central cooled annular	4-10	1375	1315	1.9	2.6
Packed-bed, annular	4-11	1780	1720	3.5	5.5
Multihole packed-bed	4-12	1625	1560	3.5	5.5

NOTE: Temperatures for all packed-bed types are based on thermal conductivity values for pyrocarbon-coated particles which are randomly packed. Temperatures for releasing fuel in similar geometries are expected to be lower than in the packed-bed types.

Table 4.6

RELATIVE FISSION-PRODUCT ACTIVITIES IN MAIN COOLANT CIRCUIT

Fuel-element Type	Total Gaseous Activity (curies) [*]	Plateout Activity After one Day Decay (curies)
Purged coated fuel particles	5×10^5	5×10^5
Unpurged coated fuel particles (with side stream trap)	1×10^6	5×10^7
Purged releasing fuel	4×10^8	5×10^7

* For a "minimum" trapping system, lower levels are obtainable. They are a function of cost.

As a means of comparing the relative merits of each fuel-element design from a fabrication cost standpoint, a summary was made of relative fabrication costs for each case. The cost summary is shown in Table 4.7. The costs included in making these comparisons consist of uranium and thorium oxide feed material, particle forming, compacting, graphite components, assembly, fuel use charges during fabrication, fuel losses during fabrication, scrap recovery, and overhead costs. The BeO spines used in some elements in Table 4.7 are based on a BeO cost of \$10.70 per pound, with no credit given for BeO recovery. The effects on costs of remote refabrication or of losses associated with reprocessing recycled fuel have not been considered in this comparison.

The work performed to date on advanced fuel elements for TARGET has shown that many variations in design may be pursued in order to develop an optimum fuel element. Considerable research and development work needs to be performed to select and develop a final fuel-element type. The packed-bed fuel element with beryllium oxide spine has been chosen at the present time as a reference design because it exhibits fairly desirable nuclear characteristics, appears feasible to fabricate by economical production techniques, operates at acceptable temperatures, and exhibits acceptable primary coolant fission-product activities.

4.3.2. Nuclear Considerations

4.3.2.1. Fuel Geometry. From the standpoint of nuclear performance, the detailed geometrical structure of the fuel element is of secondary importance. With one significant exception, the performance characteristics depend upon the quantity of each material present in the fuel and not on its spatial arrangement. The exception is thorium, which is the major resonance absorber subject to self-shielding. In the packed-bed fuel element, as opposed to the Peach Bottom type, the effective thorium cross section is reduced because of the greater heterogeneity of the fuel design. Slightly more thorium is therefore required in order to achieve the same reaction rate relative to the uranium. The effect is not large enough to significantly change the conversion-ratio-fuel-cost relationships established in the survey described below which was based on the Peach Bottom fuel concept. The labelling of the results (by moderator-thorium atomic ratio) would, of course, have to be changed in order to correspond to the packed-bed fuel.

Fuel dimensions are considered appropriate to a mean power density of 7.0 kw/liter for all cases.

Table 4.7

COMPARISON OF FUEL FABRICATION COST

<u>Fuel Element Design</u>	<u>Cost Factor (Peach Bottom Type = 1.0)</u>
<u>Retaining Fuel with Graphite Spine:</u>	
Enlarged Peach Bottom (Fig.4.12)	1.0
Nonpurged (Fig. 4.13)	0.8
Packed-bed, slotted body (Fig. 4.14)	0.8
Central cooled annular (Fig. 4.15)	1.3
<u>Retaining Fuel with Beryllium Oxide Spine:</u>	
Enlarged Peach Bottom (Fig. 4.12)	1.5
Nonpurged (Fig. 4.13)	1.3
Packed-bed, annular (Fig. 4.16)	1.4
Multihole (Fig. 4.17)	1.3
<u>Releasing Fuel with Graphite Spine:</u>	
Multihole graphite (Fig. 4.18)	0.7
Graphite body in Sleeve (Fig. 4.19)	0.9
Annular matrix (Fig. 4.20)	0.7
<u>Releasing Fuel with Beryllium Oxide Spine:</u>	
Multihole (Fig. 4.17)	1.0
Annular matrix (Fig. 4.21)	1.1

4.3.2.2. Fission-product Retention. The noble-gas fission products are of particular importance as neutron absorbers. The most significant ones are Kr^{85} , Xe^{131} , Xe^{135} , and Xe^{133} (the last being a long-lived precursor of Cs^{133}). This group is typically responsible for 3% to 4% of the total neutron absorptions in the TARGET core and represents about half of the total fission-product absorptions in a 6-year cycle, more than half with a shorter life. They are also relatively poorly retained in ceramic fuels. Two extreme assumptions were made with respect to this retention, namely that the release fraction was either zero or 100%. Actual fuels will fall between these limits; fuels with coated particles show noble gas release fractions of a few percent, those with uncoated particles 50%-100%.

4.3.2.3. Spine Composition. The central spine of the fuel element is a nonstructural member which must, however, withstand the maximum temperatures achieved in the fuel element (e. g., 1500°C) without disintegration or change of phase. The three most suitable materials for these conditions from the nuclear standpoint are, in decreasing order of merit, beryllium carbide, beryllium oxide, and graphite. The first two are closely similar, however, and the oxide is at present preferred on the basis of its physical characteristics and level of development as a reactor material. Graphite, although inferior from the nuclear standpoint, is far less costly than BeO and was retained in the survey as a standard of comparison. Lithium release from the BeO is a possibility, but was not assumed in the survey.

4.3.2.4. Fuel Cycle. In preliminary explorations, two types of uranium recycle were identified as particularly advantageous. In both types, the fuel consists of a "thorium phase," which may contain some uranium, and a uranium-only phase, which are separated, or are separable, from each other following discharge from the core. The uranium phase is then exported (presumably for use in fast-spectrum or burner reactors). The makeup feed (93.5% U^{235} was assumed) is included with the uranium phase and is thus irradiated for only one cycle. In this way the deleterious isotope U^{236} is not allowed to accumulate as it would if all uranium were to be recycled repeatedly. The uranium phase is burned to about 95% in a six-year cycle and has a final enrichment level of the order of 25%.

The two methods considered differ in their treatment of the uranium (mostly U^{233}) produced in the thorium phase. In the first method, which was chosen for the survey, the U^{233} is returned to the core with the makeup feed in the uranium phase and is therefore irradiated for only one cycle following the one in which it was produced. In the second

method, the U^{233} is repeatedly recycled with the thorium phase, and the uranium phase consists only of the enriched uranium feed material. Some U^{236} is in this method recycled back to the core; but since this U^{236} must be formed by four successive radiative captures from thorium, it is small in quantity. Active isotope U^{232} is also recycled in this method, but this is probably preferable to exporting it; in the first method, it reduces the market value of the exported uranium rather drastically.* The first method is very slightly superior in conversion ratio for 6-year cycles, but the difference is not significant, and for 3-year or 4-year cycles the first method is not optimum. The exported material is, for these short cycles, superior to the makeup feed with respect to neutron economy, and the conversion ratio could be improved by increasing the recycle stream and reducing the makeup feed as in the second method. The first method has finally an apparent advantage in that the thorium phase, which accounts for most of the fuel volume, is not contaminated with recycled uranium during the particle preparation process; the radioactivity and criticality problems are therefore restricted to a small fraction (3% to 4%) of the total particles. In all cases, the fuel cycles calculated were at the ultimate equilibrium state.

4.3.2.5. Fuel Lifetime and Cycle Time. The fuel lifetime is an essentially arbitrary variable. In general, the fuel-fabrication costs are such that lifetimes of less than 4 years are comparatively uneconomic; on the other hand, achievable conversion ratio drops rapidly for lifetimes greater than 6 years, without much compensating reduction in fuel cycle cost. The cycle time, or reload period, is also arbitrary. The ratio of the cycle time to the fuel lifetime is the fraction of the core which is reloaded at the beginning of each cycle. This was assumed to be one-sixth in the survey. This provides for annual reloading with a 6-year fuel life, or a 10-month reloading interval with a 5-year life, etc. The 6-year life has been chosen as a convenient point of reference, but this does not imply that a different life may not be preferable for reasons of fuel performance, economics, or conversion capability. The effect of refueling more frequently than once a year has been studied, and a significant improvement in conversion ratio has been found. For a six year fuel residence time, the conversion ratio was increased approximately 0.015 by refueling semiannually rather than annually.

4.3.2.6. Nuclear Data. The validity of any nuclear analysis is limited by that of the data assumed in the calculations. In the case of the thorium fuel cycle, the thermal neutron cross sections of the more important nuclides are thought to be reasonably well known. There is still considerable uncertainty with respect to epithermal cross sections, however, and the situation is especially serious with respect to the

*A schedule of price deductions based upon U^{232} content is shown in the Federal Register, May 29, 1963, p. 5315. These deductions are not included in the simplified fuel cost calculations shown in this report.

epithermal capture-fission ratio in U^{233} . The values currently in use at General Atomic (referred to as normal GA cross sections) are based upon a multilevel fit by K. V. Moore, et al. (IDO-16394, 1957, p. 34), in the 1 to 10 ev range; and upon fission cross sections reported by L. G. Miller, et al. (IDO-16373, 1957, p. 27), in the 10-1000 ev range. Capture-fission ratios lower by about 20% were obtained in an integral measurement by Halperin, et al. (Nuclear Sci. and Eng., Vol. 16, pp. 245-7, 1963), at Oak Ridge. The survey was therefore repeated with a U^{233} cross-section set modified to conform to the ORNL measurement (referred to as GA modified cross sections). Significant uncertainties in cross sections also exist for thorium, but these are of less concern since they affect only the specification of fuel composition, rather than the achievable performance characteristics of the whole fuel cycle. Finally, epithermal cross sections for fission products are not very reliable; predictions of fuel life for a given loading are, for this reason, particularly questionable, even when initial reactivity is known (as from critical experiments).

Economics Basis. A simplified cost calculation was employed in the survey, in which fuel is either AEC-owned, with 4.75% use charge on "fed" uranium only, or privately owned, with a 10% inventory charge on all fuel. No working capital was included. Three estimated fuel-fabrication costs were assumed for each fuel type, which for simplicity were assumed to be flat rates, independent of fuel type or loading, applicable to a full core. For graphite-spine fuel elements, core fabrication costs were assumed to be \$25,000,000, \$18,000,000, and \$10,000,000. For BeO spine fuel elements, the fabrication costs were increased by 22%, giving \$30,500,000, \$22,000,000, and \$12,200,000 respectively. The high figures correspond approximately to present costs, the intermediate figures to near-term future cost projections, and the low figures to future cost projections assuming an improved and more fully developed fabrication process; they are, however, to be regarded only as rough guides.

Survey Results. In Figs. 4.22 through 4.27 some of the results of the survey are shown. The results shown are the "best" so far obtained; for the GA modified U^{233} cross sections, the heaviest loadings included ($C/Th = 140$ for graphite spines, $Be/Th = 24$ for BeO spines) are not yet optimum. These loadings are, however, about the heaviest possible for the fuel designs under consideration in the project and are essentially optimal for the survey assuming the original General Atomic cross-section set.

The cases with retention of volatile fission products, which are directly applicable to the selection of the reference design, are, as discussed above, conservative with respect to achievable conversion ratio. Both lithium and Xe^{131} may be released from the reference fuel

Fig. 4.22--Conversion ratio versus fuel cost with GA modified cross sections, fission products released, AEC fuel

Fig. 4.23--Conversion ratio versus fuel cost with GA cross sections, fission products released, AEC fuel

Fig. 4.24--Conversion ratio versus fuel cost with GA modified U^{233} cross sections, fission products retained, AEC fuel

Fig. 4.25--Conversion ratio versus fuel cost with modified U^{233} cross sections, fission products retained, private fuel

Fig. 4.26--Conversion ratio versus fuel cost with GA cross sections, fission products retained, AEC fuel

Fig. 4. 27--Conversion ratio versus fuel cost with GA cross sections, fission products retained, private fuel

to a significant extent, and with full optimization of the fuel management method, conversion ratios above 1.00 with GA modified U^{233} cross sections, or 0.96 with the General Atomic cross sections, appear to be achievable for the 6- and 4-year lifetimes. For a 6-year lifetime, the conversion ratio is seen to be lower by about 4%, but the fuel cost is lower by 0.1 to 0.3 mills.

Additional data are presented below in Tables 4.8 through 4.14.

4.3.3. Helium Purification System for Releasing Fuel

A very preliminary design of a "minimum" helium purification system to cope with releasing fuel was made. This system would result in a main loop activity of 450 megacuries and chemical impurity levels comparable to those with retaining fuel. However, the additional requirement of maintaining a minimum carbon monoxide level was placed on the system. The system consists of

1. Pancake traps which consist of layers of charcoal interspersed between extended surface area heat exchangers,
2. A chemical impurity removal system,
3. A batch activity cleanup system for maintenance,
4. A cooling water system, and
5. A system which would supply purified purge helium to the control-rod drives.

The pancake traps remove from the purge stream the short-lived noble-gas activity that decays to long-lived plateout activity. The design of these beds results in a high decay-heat removal capability. Following these beds the purge stream is returned to the main loop.

A side-stream chemical system (handling 500 lb/hr) would remove the chemical impurities from the main loop. This system uses a copper oxide oxidizer to convert hydrogen and carbon monoxide to water and carbon dioxide. Subsequently, molecular sieve beds remove this water and carbon dioxide.

A small portion of the preceding stream (35 lb/hr) is processed in a low-temperature bed to remove nitrogen and some long-lived noble-gas activity. This system is also used as a batch-activity cleanup system for the main loop before maintenance operations are performed.

Another side stream system consisting of two or three low-temperature charcoal beds, which are regenerated into the main loop following short adsorption cycles, supply the required control-rod drive purge gas.

The helium purification system as described here is so costly as to make the releasing fuel case less attractive than the retaining fuel. This fact strongly points up the need for additional work in fission product removal methods.

Table 4.8

TOTAL ASSUMED LOADINGS OF MODERATOR AND THORIUM

Atomic Ratio	Thorium* (tonnes)	BeO (tonnes)	Carbon (tonnes)
Be/Th = 40	60.43	312	358.3
= 36	67.15	312	358.3
= 32	75.54	312	358.3
= 28	86.33	312	358.3
= 24	100.73	312	358.3
C/Th = 230	41.38	-----	492.0
= 200	47.59	-----	492.0
= 170	55.98	-----	492.0
= 140	67.98	-----	492.0

*The values for thorium are full-core, "initial" loadings and are equal to six times the feed required per cycle. The number of fuel elements corresponding to these loadings is 5737, which is higher than the reference core number (5489) by 4.5%.

Table 4.9

INITIAL Be/U RATIOS FOR EQUILIBRIUM CYCLES

Assumptions	Be/ Th	3-Yr. Life	4-Yr. Life	5-Yr. Life	6-Yr. Life	7-Yr. Life	8-Yr. Life
GA cross sections, F. P. * retained	36	1319	1213	1096	981	----	----
	32	1175	1095	999	908	831	752
	28	----	983	916	840	770	702
Modified cross sections, F. P. * retained	36	1546	1349	1228	1010	----	----
	32	1286	1224	1142	1051	963	873
	28	1197	1164	1100	1024	945	867
	24	915	890	850	800	746	695
GA cross sections, F. P. * (volatile) released	36	1630	1565	1475	1365	----	----
	32	1427	1388	1328	1250	1169	1077
	28	1256	1230	1178	1111	1040	973
Modified cross sections, F. P. * (volatile) released	36	1930	1905	1815	1693	----	----
	32	1583	1596	1564	1515	1458	1372
	28	1466	1491	1473	1431	1380	1330
	24	1096	1105	1095	1055	1010	965

*F. P. = fission products.

Table 4.10

 U^{235} FEED REQUIRED PER CYCLE* (Kg U^{235})

Assumptions	Be/ Th	3-Yr. Life	4-Yr. Life	5-Yr. Life	6-Yr. Life	7-Yr. Life	8-Yr. Life
GA cross sections, F. P. [†] retained	36	102	105	125	156	----	----
	32	122	118	133	157	186	226
	28	----	146	151	171	199	236
Modified cross sections, F. P. [†] retained	36	69	74	89	113	----	----
	32	93	81	86	103	127	161
	28	106	84	83	94	114	141
	24	148	123	107	106	116	133
GA cross sections, F. P. [†] (volatile) released	36	41	32	35	48	----	----
	32	58	40	37	43	56	78
	28	87	62	56	62	74	91
Modified cross sections, F. P. [†] (volatile) released	36	16	4	5	15	----	----
	32	31	5	0	0	0	8
	28	41.7	9.8	0	0	0	0
	24	83	35	6	0	0	0

*The cycle time is one-sixth of the fuel life.

[†]F. P. = fission products

Table 4.11

NET FISSILE URANIUM CONSUMPTION PER CYCLE* (Kg U)

Assumptions	Be/ Th	3-Yr. Life	4-Yr. Life	5-Yr. Life	6-Yr. Life	7-Yr. Life	8-Yr. Life
GA cross sections,	36	29	47	76	112	----	----
F. P. [†] retained	32	24	30	66	97	132	175
	28	----	24	62	90	122	163
Modified cross sections,	36	15	27	50	83	----	----
F. P. [†] retained	32	9	17	34	59	80	124
	28	5	11	21	40	64	95
	24	5	6	11	23	41	64
GA cross sections,	36	-10	-5	6	23	----	----
F. P. [†] (volatile)	32	-12	-11	-4	8	26	50
released	28	-5	-7	0	14	30	51
Modified cross sections	36	-20	-21	-15	-2	----	----
F. P. [†] (volatile)	32	-28	-35	-31	-25	-20	-12
released	28	-29	-39	-38	-31	-27	-24
	24	-28	-43	-52	-48	-42	-37

*The cycle time is one-sixth of the fuel life.

[†]F. P. = fission products.

Table 4. 12

FISSIONS PER INITIAL FISSILE ATOM (FIFA)

Assumptions	Be/ Th	3-Yr. Life	4-Yr. Life	5-Yr. Life	6-Yr. Life	7-Yr. Life	8-Yr. Life
GA cross sections, F. P* retained	36	1. 192	1. 471	1. 663	1. 785	----	----
	32	1. 062	1. 328	1. 516	1. 653	1. 751	1. 811
	28	----	1. 184	1. 380	1. 519	1. 624	1. 691
Modified cross sections, F. P. * retained	36	1. 397	1. 625	1. 851	2. 107	----	----
	32	1. 161	1. 475	1. 720	1. 898	2. 032	2. 106
	28	1. 081	1. 403	1. 657	1. 851	1. 994	2. 091
	24	0. 885	1. 167	1. 411	1. 612	1. 770	1. 895
GA cross sections, F. P. * (volatile) released	36	1. 474	1. 900	2. 238	2. 483	----	----
	32	1. 290	1. 685	2. 008	2. 275	2. 464	2. 596
	28	1. 135	1. 482	1. 775	2. 007	2. 192	2. 344
Modified cross sections, F. P. * (volatile) released	36	1. 753	2. 294	2. 737	3. 059	----	----
	32	1. 430	1. 923	2. 356	2. 738	3. 075	3. 307
	28	1. 325	1. 797	2. 220	2. 586	2. 907	3. 200
	24	1. 068	1. 471	1. 860	2. 191	2. 487	2. 752

*F. P. = fission products

Table 4. 13

NEUTRON YIELD PER FISSILE ABSORPTION (η)

Assumptions	Be/ Th	3-Yr. Life	4-Yr. Life	5-Yr. Life	6-Yr. Life	7-Yr. Life	8-Yr. Life
GA cross sections, F. P. * retained	36	2. 159	2. 161	2. 156	2. 148	----	----
	32	2. 149	2. 152	2. 149	2. 143	2. 135	2. 126
	28	----	2. 137	2. 137	2. 132	2. 124	2. 116
Modified cross sections, F. P. * retained	36	2. 207	2. 201	2. 197	2. 194	----	----
	32	2. 190	2. 197	2. 196	2. 191	2. 184	2. 175
	28	2. 188	2. 200	2. 201	2. 197	2. 190	2. 183
	24	2. 173	2. 183	2. 191	2. 190	2. 186	2. 181
GA cross sections, F. P. * (volatile) released	36	2. 190	2. 193	2. 189	2. 183	----	----
	32	2. 149	2. 185	2. 184	2. 179	2. 172	2. 166
	28	2. 160	2. 168	2. 169	2. 165	2. 159	2. 153
Modified cross sections, F. P. * (volatile) released	36	2. 237	2. 239	2. 235	2. 229	----	----
	32	2. 178	2. 231	2. 229	2. 225	2. 222	2. 217
	28	2. 220	2. 233	2. 235	2. 229	2. 225	2. 222
	24	2. 199	2. 222	2. 227	2. 226	2. 223	2. 219

*F. P. = fission products.

Table 4. 14

CONVERSION RATIO

Assumptions	Be/ Th	3-Yr. Life	4-Yr. Life	5-Yr. Life	6-Yr. Life	7-Yr. Life	8-Yr. Life
GA cross sections, F. P. * retained	36	0.936	0.918	0.896	0.871	-----	-----
	32	0.946	0.930	0.910	0.889	0.869	0.847
	28	-----	0.930	0.916	0.898	0.878	0.858
Modified cross sections, F. P. * (volatile) released	36	0.966	0.954	0.931	0.909	-----	-----
	32	0.981	0.971	0.954	0.934	0.912	0.889
	28	0.992	0.989	0.973	0.956	0.935	0.916
	24	0.994	0.994	0.986	0.974	0.959	0.943
GA cross sections, F. P. * (volatile) released	36	1.023	1.012	0.995	0.975	-----	-----
	32	1.029	1.022	1.009	0.992	0.974	0.954
	28	1.017	1.013	1.002	0.987	0.970	0.955
Modified cross sections, F. P. * (volatile) released	36	1.050	1.040	1.025	1.005	-----	-----
	32	1.069	1.068	1.055	1.041	1.028	1.011
	28	1.073	1.074	1.065	1.051	1.038	1.027
	24	1.070	1.079	1.079	1.069	1.057	1.045

*F. P. = fission products.

4.4. STEAM CYCLES

4.4.1. Throttle Temperature and Pressure

The effect of the throttle and reheat steam conditions on the steam cycle performance has been calculated. The results are presented in Fig. 4.28.

The graph shows the plant heat rates plotted versus the steam turbine throttle pressure for various combinations of throttle and reheat temperatures. The following steam cycles were evaluated.

1. Single and double gas reheat.
2. Double steam reheat.

The term gas reheat is applied to steam cycles in which the exhaust steam from the high-pressure turbine is returned to the steam generator and is reheated by the primary coolant before entering the intermediate pressure turbine.

Double gas reheat means that the exhaust steam from the intermediate-pressure turbine exhaust is similarly returned and reheated by the hot helium before entering the low-pressure turbine.

The term double steam reheat is applied to steam cycles in which the reheating of the exhaust steam from the high-pressure and intermediate-pressure turbines is accomplished in the shell side of steam-to-steam heat exchangers. The tube side of the steam-to-steam exchangers is fed live steam from the steam generator superheater outlet. After the live steam is cooled in the reheaters, it is returned to the steam generators where it is re-superheated before entering the high-pressure turbine.

The heat rates shown are net station heat rates--which means they include all losses incidental to the operation of the nuclear steam raising system and the steam cycle, including the total station auxiliary power.

The reactor thermal power in all cases is nominally 2270 Mw(t), resulting in differing net station send out commensurate with the respective heat rates. The comparison curves of Fig. 4.28 are shown for the case of a turbocirculator and will be negligibly different for a steam-turbine-driven circulator.

Fig. 4.28--The effect of steam conditions on performance; FFT = final feedwater temperature, T_{TH} = throttle temperature, and T_{RH} = reheat temperature

A seven-state regenerative feedwater system has been assumed with top heater extraction from first cold reheat point. The optimum reheat pressure is mainly a function of the throttle pressure. Final feedwater temperatures (FFT) are shown on Fig. 4.28 for three representative throttle pressures: 2400, 3500, and 4500 psig.

Several general observations can be derived from this curve:

1. The throttle pressure and the throttle and reheat temperature strongly affect the plant efficiency.
2. The effect which the throttle temperature has on the efficiency increases with increasing throttle pressure.
3. Double reheat becomes more advantageous with higher throttle pressures.
4. Double steam reheat is insignificantly less efficient than double gas reheat. The gas reheat becomes more efficient with higher throttle pressures.

A parametric plant design study has not been attempted at this time to determine the optimum throttle and reheat temperatures. For the comparative studies, the throttle and reheat temperatures have been limited to 1050°F for all steam cycles. The potential of higher steam temperatures is nevertheless indicated on Fig. 4.28 by the performance curves of double steam and gas reheat for 1100F/1050F/1050F steam conditions. The large steam turbine manufacturers in the United States advised that these would be the temperature ceilings for throttle and reheat temperature for large steam turbines for some time to come. The highest steam temperature level presently used for continuous operation in fossil-fired power plants is 1150°F. The high cost of austenitic steel and other technical problems associated with this material has established a trend to lower steam temperatures in the electric utility industry. These factors have been taken into consideration in the selection of TARGET steam conditions.

By a general hand rule the tube metal temperature is approximately 50°F higher in a convective steam generator as compared to radiant heat transfer equipment used in fossil-fired plants for a given throttle temperature. This would limit steam temperature from a TARGET steam generator to 1100°F in comparison with the 1150°F from a fossil-fired plant. As the experience with metals at these elevated temperatures at high pressures and large flow rates is very scarce, it was considered prudent

to remain 50°F below the equivalent maximum attainable steam temperature of 1100°F.

The maximum steam temperature of 1050°F is applied to the throttle temperature in the case of gas reheat. In the case of steam reheat, the temperature from the steam reheater is 1000°F; no attempt has been made at this time to optimize the terminal temperature difference of 50°F at the steam reheaters.

The trend of the power cost as a function of the throttle pressure as determined by these comparative studies results from a comparison of Case 1 and Case 10. Both cases employ a double steam reheat cycle. The throttle temperature in both cases is 1050°F and the reheat temperatures 1000°F. The throttle pressure is 3500 psig in Case 1, and 4500 psig in Case 10. The performance of Cases 1 and 10 are given in Table 4.15.

The heat balances for both cases are presented in Figs. 4.29 and 4.30.

The cost comparison indicates that the power cost rises with increasing throttle pressure. This increase is 3% for a throttle pressure change from 3500 to 4500 psig. This trend will also prevail for a single gas reheat cycle, although the power cost differential will be considerably less, due to the fact that a much smaller portion of the total construction cost is allocated to the high-pressure system.

The throttle pressure comparison is only applicable for the specific throttle and reheat temperature conditions. Figure 4.28 also shows the increasing effect which the throttle pressure has on the steam cycle efficiency with higher throttle temperatures. A 1000-Mw(e) HTGR power plant will optimize at higher throttle pressures with increasing throttle and reheat temperature conditions.

4.4.2. Steam Versus Gas Reheat

The application of a double gas reheat steam cycle for TARGET was precluded from the study after an initial investigation. With double gas reheat the second reheat pressure optimizes below 300 psia, which is significantly below the primary system pressure of 450 psia. The pressure differential would always be in the direction of inleakage of primary coolant system activity into the secondary coolant system, should any loss of reheat system integrity occur. With single gas reheat,

Table 4. 15

PERFORMANCE OF CASES 1 AND 10

	Case 1	Case 10
Reactor thermal output, Mw(t)	2270.00	2269.50
Gross generator output, Mw	1086.62	1101.00
Boiler feed pump power, Mw	27.21 + 9.45	34.09 + 11.83
Other auxiliary power, Mw	10.91	10.97
Net station send out, Mw	1066.26	1078.20
Net station heat rate, Btu/kw-hr	7266	7184
Net station efficiency, %	46.97	47.51
Turbine arrangement	CCQF 52 in	CCQF 52 in
Power cost, mill/kw-hr	3.77	3.86
Specific capital cost, \$/kw	129.61	167.70

steam pressure is always above primary system pressure at or near full-load operating conditions. The steam raising equipment for the single gas reheat cycle is described in detail in Section 3.4.

The steam generator sections are similar in design to those for the reference design (see Section 3.4). The helium heated reheater bundle is replaced by a resuperheater bundle. Since the resuperheater bundle must be designed for 3500 psi throttle pressure, whereas the reheater bundle is designed for 533 psi (external), this bundle will be more costly.

The steam reheaters are large shell-and-tube heat exchangers with two exchangers used in parallel for the first reheat and two for the second. Each exchanger consists of a carbon steel shell, which is cooled by the incoming reheat steam, and a stainless steel tube and header system for the 4000 psi steam. The tubes are bent to form serpentine panels with a single inlet header at the center of the unit and an outlet header at each end. The pertinent design features of the reheaters are shown in Table 4.16 for a single unit.

The results of the detailed cost estimates for Cases 1 and 4 are given in Table 4.17.

The performance of the double steam reheat and single gas reheat steam cycles are compared in Case 1 and Case 4 (see Table 4.17). These data given in the Table show that the application of steam reheat is not economical as compared to gas reheat. The total power costs are approximately 6% greater for the steam reheat case.

Fig. 4.29--Heat balance diagram
for double steam reheat
(Case 1)

Fig. 4.30--Heat balance diagram
for double steam reheat
(Case 10)

Table 4. 16
PERTINENT DESIGN FEATURES FOR STEAM REHEATERS

	First Reheat (1050 psia)	Second Reheat (300 psia)
Number of exchangers	2	2
Surface area, ft ²	33,000	35,000
Length of exchanger, ft	54	61
Diameter of exchanger	9 ft 1 in.	9 ft 9 in.
Dry weight of exchanger, tons	212	170

Table 4. 17
PERFORMANCE OF CASES 1 AND 4

	Case 1	Case 4
Nominal reactor thermal output, Mw	2270.00	2270.00
Gross generator output, Mw	1086.62	1074.74
Boiler feed pump power, Mw	27.21+9.45	25.61+8.89
Other auxiliary power, Mw	10.91	11.21
Net station send out, Mw	1066.26	1054.37
Net station heat rate, Btu/kw-hr	7266	7347
Net station efficiency, %	46.97	46.45
Turbine arrangement	CCQF 52 in.	CCQF 52 in.
Turbine throttle flow, 10 ⁶ lb/hr	6.456	6.637
Power cost, mill/kw-hr	3.77	3.56
Specific capital cost, \$/kw	129.61	118.68

4. 4. 3. Dry Cooling Tower

The application of a dry cooling tower system (Case 11) for heat rejection from the steam plant has been investigated (see Fig. 4. 31). A dry cooling tower greatly lessens the problem of site selection, since it virtually eliminates the requirement for water supply.

The dry cooling tower system employs a single shell-injection type spray condenser in place of a conventional surface condenser. Other components included in the system are circulating pumps, cooling elements, tower fans, recovery turbine, air ejectors, and water storage tank.

Water of condensate purity from the cooling towers enters the condenser sprays and condenses the steam from the turbine. The mixture of condensate and circulating water is withdrawn from the condenser by means of circulating water pumps. That part of the circulating water pump discharge flow required by the feedwater system is returned to the condensate pumps; the largest portion of the flow is directed to the cooling elements which are arranged in the towers. Water recovery turbines are installed in the tower return piping near the condenser.

An initial comparison between natural and mechanical-draft cooling towers led to the selection of an induced-draft fan tower arrangement. The high investment cost for the construction of natural-draft towers eliminated this type from further investigation.

With this design, which is based on information obtained from English Electric Corporation, the cooling of the circulating water is accomplished in two mechanical-draft towers. Each tower is divided into four sections on both the air and water sides for flexibility of operation. Any of the sections can be used independently, but all sections are required to be in operation for full-load design conditions. Each tower is 46 ft high, 80 ft wide, and 1280 ft long. There are 64 fans per tower. The fans are 34 ft in diameter and are driven by 150 hp motors (see Fig. 4.32).

The circulating water system is designed so that the entire circuit between the circulating water pumps and recovery turbines is above atmospheric pressure. The point of minimum pressure is at the top of the cooling water elements. Maintaining an above-atmospheric pressure at the top of the coolers would provide an excess of pressure at the condenser spray nozzles and the water turbines are used as an economical means of recovering a portion of the excess head.

The dry cooling tower system is designed to produce a 2.0 in. Hg back pressure with 62.5°F ambient air temperature. Four circulating water pumps driven by 6000 hp motors are designed for a total flow of 668,000 gpm with a developed head of 55 psi. Gross circulating pump power is 28,993 bhp. The recovery turbines account for 3679 bhp. The net circulating pump power is 25,314 bhp.

The results of the detailed cost estimates (see Case 1 and Case 11 on Table 4.1) are as follows:

	Open cooling water (Case 1)	Dry cooling tower (Case 11)
Power cost, mill/kw-hr	3.77	4.56
Specific capital cost, \$/kw	129.61	167.70

Fig. 4.31--Heat balance diagram
for dry cooling tower
(Case 11)

Fig. 4.32--View of Mechanical-draft dry cooling tower

The dry cooling tower system has not been optimized, and a significant cost reduction appears possible for this system. Nevertheless, the above data indicate that a dry cooling tower system can only be justified over an open cooling water system when the water cost for the circulating water makeup is in the range of \$0.60 to \$0.70 per thousand gallons. However, the high efficiency of the TARGET system does make the use of a dry cooling tower system less of an economic penalty than would be associated with a plant with a lower thermodynamic efficiency.

4.5. COST COMPARISONS

From the studies that have been made of various alternative designs, costs have been prepared for the complete power plant. During the course of the preparation of the cost estimates, it became clear that certain of the selected designs could not possibly be economic. These were therefore eliminated without taking their costing to the final figure.

Estimates have been prepared by the same methods as given in Section 3.16 with the following additions:

Steel reactor vessel and containment costs were based on preliminary quotations received from Chicago Bridge and Iron Company.

Interest during construction for steel reactors is 10.7% (48 months at 6%). Interest during construction for concrete reactors is 9.4% (42 months at 6%).

All plants have the same reactor heat output, and therefore net sendout varies with the efficiency. All plants are identical, except for pressure vessel, circulator, reheat, and fuel type as listed in Table 4.18.

Table 4. 18

TARGET COST COMPARISONS

	Case 1	Case 1A	Case 2	Case 3	Case 4	Case 5	Case 5B	Case 6	Case 10	Case 11
Pressure vessel	Concrete	Concrete	Steel	Concrete	Concrete	Steel	Steel	Steel	Concrete	Case 1 with dry cooling tower
Circulator	Gas	Gas	Steam	Steam	Gas	Gas	Gas	Gas	Gas	
Reheat	Double steam	Double steam	Single gas	Single gas	Single gas	Double steam	Double steam	Single gas	Double ^a steam	
Fuel type	Retain-ing	Releas-ing	Retain-ing	Retain-ing	Retain-ing	Retain-ing	Releas-ing	Retain-ing	Retain-ing	
Capital cost, \$1000	138,160	144,170	140,310	123,240	125,090	153,790	163,380	140,920	147,650	168,710
Net sendout, Mw(e)	1066	1066	1032	1040	1054	1066	1066	1054	1078	1006
Specific capital cost, \$/kw	129.61	135.24	135.96	118.50	118.68	144.27	153.26	133.70	136.97	167.70
Capital charge at 14%, mill/kw-hr	2.59	2.69	2.70	2.37	2.37	2.87	3.06	2.67	2.69	3.35
Insurance charge, mill/kw-hr	0.04	0.04	0.04	0.04	0.04	0.04	0.04	0.04	0.04	0.04
Operation and maint., mill/kw-hr	0.20	0.20	0.20	0.20	0.20	0.20	0.20	0.20	0.20	0.20
Working capital \$22x10 ⁶ core cost, mill/kw-hr	0.16	0.14 ^b	0.16	0.16	0.16	0.16	0.14 ^b	0.16	0.16	0.16
Fuel cycle cost (conv. ratio 0.97)	0.78	0.52	0.81	0.80	0.79	0.78	0.52 ^b	0.79	0.77	0.83
TOTAL POWER COST	3.77	3.59 ^b	3.91	3.57	3.56	4.05	3.96 ^b	3.86	3.86	4.58

^a 4500 psi.^b Conversion ratio 1.06, fabrication cost \$19 x 10⁶