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EVALUATION OF THORIUM FUELS
FOR THE HEAVY WATER ORGANIC
COOLED REACTOR

July 1966

Atomic Energy Division
The Babcock & Wilcox Company
Lynchburg, Virginia

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for the
HEAVY WATER ORGANIC COOLED REACTOR

Approved by W. M. Vannoy, Jr.
Project Manager

July 1966

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THE UNITED STATES ATOMIC ENERGY COMMISSION
by
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ABSTRACT

This report summarizes the results of work done by The Babcock & Wilcox Company to evaluate thorium fuels for the Heavy Water Organic Cooled Reactor (HWOCR). This work is a part of the USAEC's overall program to develop economical heavy water reactors having optimum fuel utilization. The long-range objective of the thorium fuel cycle development is the proof of performance of thorium fuel elements in an HWOCR environment.

The primary effort to date has been related to the selection of a thorium fuel element for future development. Three types of fuel elements were studied in detail: thorium metal nested cylinders, thorium oxide clustered pins, and thorium monocarbide clustered pins. On the basis of extensive parameter studies an optimum design of each type was selected for final evaluation. The economics and fuel utilization of the three fuel element types were evaluated for use in a 1000-MWe HWOCR. Technical feasibility and development costs were also considered in the selection process. A test program was planned in conjunction with the optimization studies to define key technical problems and the development required.

On the basis of the comparative evaluation it is recommended that both the oxide and the metal fuel elements be carried through several phases of irradiation testing before a final selection is made. Priority should be given to the development of the ThO_2 fuel element. The initiation of test work as soon as it is practical will give substance to the continued effort toward achieving the long-range objectives of the HWOCR Program.

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1. SUMMARY

The primary effort of the Commission's Heavy Water Program, under a contract with AI-CE, is the development of the HWOCR to operate on the uranium fuel cycle. A secondary effort, under a contract with B&W, is the development of the thorium fuel cycle for the HWOCR. At minimum incremental R&D expense, the HWOCR will have the fuel cycle flexibility to enable the efficient use of our thorium resources when technical and economic factors favor the integration of the thorium fuel cycle into the nuclear economy. Adaptation of thorium fuel to the AI-CE design will also permit proof of performance in an HWOCR environment and/or the near-term construction of a Thorium HWOCR for special applications, such as the development of a nuclear power industry in a nation having large thorium resources.

The HWOCR Thorium Program overall plan has two major phases: Phase I, Evaluation and Planning; Phase II, Thorium Fuel Cycle Development. This report summarizes the results of work done in Phase I, which covers the period of January 1965 through June 1966. The objectives of the Phase I work program were to select a reference thorium fuel element and to prepare a test program plan for the development of this fuel element under Phase II. Three basic types of fuel elements were studied:

<u>Thorium form</u>	<u>Fuel cladding</u>	<u>Geometry</u>
Metal	Zircaloy	Nested cylinders
Oxide	SAP	Clustered pins (finned)
Monocarbide	SAP	Clustered pins (finned)

The economics and fuel utilization of these fuel elements were evaluated for a 1000 MWe thorium-optimized HWOCR and for the 1000 MWe AI-CE design. The comparison of the thorium-optimized reactor and

the thorium-fueled AI-CE reactor shows that the two systems are basically compatible. The design of the thorium-optimized reactor was carried out as an independent effort by B&W. In such areas as refueling and pressure tube design there were basic differences between the uranium and thorium designs. However, since compatibility was demonstrated, the AI-CE reactor arrangement was adopted for the thorium core.

The evolution of a near-optimum design for each type of fuel element involved the evaluation of a number of designs encompassing a broad range of parameters, which significantly affect power costs and fuel utilization. ORNL evaluated thorium metal and thorium oxide designs submitted in November 1965. This evaluation confirmed that more extensive optimization would be required before a valid comparison with alternate concepts could be made. The information developed by ORNL proved valuable to B&W in continuing the design optimization. The optimization studies led to the 1000 MWe Thorium HWO CR designs described in section 2. A design data summary is given in Table 1-1.

The relative performance of the 1000 MWe thorium-optimized HWO CR's is indicated by the following summary:

1000 MWe HWO CR conceptual design	Costs, mills/kwhr				Conversion ratio (equilibrium cycle)
	Fuel		Power		
	Investor- owned utility	Publicly- owned utility	Investor- owned utility	Publicly- owned utility	
	Investor- owned utility	Publicly- owned utility	Investor- owned utility	Publicly- owned utility	
Thorium metal	1.19	0.94	3.54	2.40	0.86
Thorium oxide	1.23	0.95	3.73	2.49	0.87
Thorium carbide	1.23	0.99	3.71	2.52	0.87

Fuel cycle costs for the three designs are about equal. The nested-cylinder design shows a slight advantage in power costs, primarily due to lower capital costs resulting from higher power per pressure tube. The cost differences are probably within the uncertainty band; therefore, factors other than economics must also be carefully evaluated.

As indicated above, the three designs have approximately the same conversion ratio. Also, the Dynamic Utilization does not show

any significant fuel utilization advantage for any design. The similar performances may indicate that economic optimization leads to about the same fuel utilization characteristics. Higher values of conversion ratio can be achieved at the expense of economics by increasing the fuel volume fraction within the pressure tube.

Each of the three fuel element types is adaptable to the AI-CE HWOCR. The ThC fuel element is identical for the thorium-optimized and the AI-CE HWOCR designs and has basically the same configuration as the 37-pin UC fuel element. The Th-U nested-cylinder fuel element is capable of more power output per unit volume than the UC fuel element; therefore, the Th-U core offers a potential "stretch" position for the AI-CE HWOCR. The ThO₂ fuel element has a lower pin heat rating (kw/ft) than the UC fuel element and requires 55 pins instead of 37 to achieve the same power output. The smaller diameter pins in the 55-pin element increase fabrication costs by about 0.1 mill/kwhr.

Fuel element test programs were planned for each type of fuel element. The key problems to be investigated for the nested-cylinder design are the fabrication tolerances of large-diameter (up to about 5 inches), thin-wall, closely packed cylinders and the dimensional stability of these cylinders under long-term irradiation (20,000 MWD/T average) in an HWOCR environment. The principal limitation on the performance of the ThO₂ fuel element is the fission gas pressure build-up in the low-ductility SAP cladding with low (approximately 300 psi or less) external pressure. There is a gross lack of data for ThC, but it is known that ThC is highly reactive with moisture in air. Before other development tasks can be undertaken, a fuel composition that is resistant to attack by moisture must be developed.

The preliminary cost estimates for the fuel element test programs (outlined in section 3) are summarized below:

Thorium metal	\$7,366,300
Thorium oxide	5,005,300
Thorium monocarbide	7,123,800

In addition, it is anticipated that a supporting experimental physics program will be conducted at SRL at an additional cost of about \$1.8 million, without much cost variation for the different fuel element types. Not estimated on an absolute basis is the development of recycle technology

required to achieve large-scale production capability. On a relative basis, it is estimated that ThO_2 would require only about one-third as much development as either Th-U or ThC.

The comparative evaluation of thorium fuels for the HWOCR is discussed in detail in section 4. On the basis of this evaluation, it was concluded that there is little incentive to develop ThC since the economic and fuel utilization performance is no better than the performance of ThO_2 and the technical obstacles to successful development of ThC are much greater. It is recommended that both Th-U and ThO_2 be carried through several phases of irradiation testing. After sufficient data have been obtained on the dimensional stability of the nested cylinders and on the fission gas limits of the SAP-clad ThO_2 pins, the correct decision can be made with a high degree of confidence. If practical considerations require the selection of a single fuel element, ThO_2 is the recommended choice. ThO_2 provides the shortest and cheapest route to the successful development of a thorium fuel element for the HWOCR. The ThO_2 design and the production processes have flexibility for adapting to changing technical and economic conditions that may favor the large-scale application of the HWOCR thorium fuel cycle.

Table 1-1. 1000 MWe Thorium HWO CR Design Data Summary

Parameter	ThO ₂ clustered-pin design	ThC clustered-pin design	Th-U nested-cylinder design
<u>Material</u>			
Fuel	ThO ₂	ThC	Th-U metal
Cladding	SAP	SAP	Zr-4
Coolant	Santowax OM, 10% HB	Santowax OM, 10% HB	Santowax OM, 10% HB
Moderator	D ₂ O	D ₂ O	D ₂ O
<u>Power</u>			
Reactor power, MWt	3100	3151	3187
Power to coolant, MWt	2936	2984	3019
Power to moderator, MWt	157	157	158
Net electric output, MWe	1076	1076	1076
Power/pressure tube, MWt (max)	4.53	6.37	10.9
Power/pressure tube, MWt (avg)	4.09	5.75	9.8
Specific power, w/gm (avg)	25.8	32.0	32.0
<u>Reactor</u>			
Number of pressure tubes	758	548	324
Pressure tube ID, in.	4.03	4.03	5.00
Pressure tube pitch, in.	9.5	11.0	11.6
Moderator tank diameter, ft-in.	26-7	26-3	21-7
Moderator tank height, ft-in.	20	20	21-4
Active core length, ft-in.	17-9.4	17-9.4	18-9
<u>Thermal-hydraulic</u>			
Steam temp, F	675	675	675
Steam pressure at turbine, psia	900	900	900
Reactor inlet temp, F	560	560	560
Reactor outlet temp, F	721.4	716.2	708
Coolant flow rate, lb/hr × 10 ⁶	107	113	119
Core pressure drop, psi	92	155	185
Surface temp excluding fouling allowances, F	830	830	820
Max heat flux, Btu/hr-ft ²	223,000	313,000	400,000
Core inlet pressure, psia	229	292	322
Core outlet pressure, psia	137	137	137
<u>Nuclear</u>			
Fuel loading (U + Th metal), kg	119,700	98,900	98,800
Avg fuel burnup, MWD/T	20,000	20,000	20,000
Initial U-235 enrichment, wt %	2.31	2.32	2.40
Equilibrium cycle conver- sion ratio	0.87	0.87	0.86
Fuel management: on-line continuous refueling, countercurrent in adjacent channels			

2. 1000 MWe THORIUM HWO CR DESIGNS

2.1. General

To provide bases for evaluating thorium fuels for HWO CR use, conceptual designs were evolved for (1) a 1000-MWe thorium-optimized HWO CR and (2) a thorium core for the uranium-optimized 1000-MWe HWO CR designed by AI-CE. A number of designs were evaluated in optimization studies that preceded the selection of a near-optimum design for each type of fuel element. (The design evolution is discussed in technical progress reports.¹⁻³) Detailed calculations have been made for the selected designs, and the resultant conceptual designs are described in this section. These designs provide the bases for defining key technical problems to be investigated in the test programs (outlined in section 3) and for the comparative evaluation of the thorium fuels (presented in section 4).

2.2. Overall Plant Design

In accordance with Commission objectives, B&W has concentrated on the reactor core and relied on the AI-CE plant design effort, with the exception of some work in the area of reactor internals to guide and support the optimization of the thorium cores. Thus the overall plant design for the thorium cores is basically identical to the HWO CR plant design described in AI-CE-MEMO-25,⁴ except that the design power is 1076 MWe, and different numbers of pressure tube assemblies are required depending on the fuel material. Operating parameters also differ slightly, but these can be adjusted to conform to the HWO CR plant values by relatively minor modifications to the fuel element design.

The reactor complex for each of the three reference thorium cores consists of pressure tube assemblies arranged vertically in a calandria vessel which contains the heavy water moderator. Each pressure tube assembly consists of a SAP pressure tube and a zirconium alloy calandria tube that surrounds the pressure tube and separates it

from the moderator, forming an insulating annulus filled with CO₂. Each pressure tube contains a stack of five fuel element assemblies. Two shield plug assemblies at each end of the fuel column hold the fuel elements in the core and prevent radiation streaming from the channel. Santowax-OM coolant enters and leaves through tee connections and "pigtail" piping which connects the process tubes to the inlet and outlet headers at the top and bottom of the reactor. The cylindrical calandria vessel contains the D₂O moderator and surrounds the fuel channel assemblies to provide a 12-inch reflector in both axial and radial directions outside of the core. The heat generated in the moderator heats the boiler feedwater. Integral end shields provide radiation protection. The concrete radial biological shield supports the reactor and end shields.

Refueling is performed with the reactor at power by two refueling machines located above and below the reactor structure. The machines are supported vertically from rail structures by carriages which position them opposite the ends of any fuel channel.

The core has one-pass, bi-directional coolant flow with bi-directional, on-power refueling. The coolant flows in opposite directions in adjacent process tubes to maintain flow in the same direction as the fuel movement. This arrangement results in lower coolant temperatures in the higher power regions of the core and thus improves thermal performance. Process tubes are orificed to match coolant flow rates to the basic radial power distribution so that the coolant temperature rise is approximately the same across all the pressure tubes.

Three primary coolant loops are provided; each consists of a turbine-driven circulating pump and evaporator, superheater, and reheater sections with interconnecting piping. At design power, organic coolant is circulated at the rate of about 115×10^6 lbs/hr, varying slightly with the different fuel materials. The pressure at the outlet mixing tank is controlled at 100 psia. Coolant is heated in the core from 560 to approximately 705 F, absorbing about 3000 MWt. The heat is used to generate superheated steam at 925 psig and 675 F and to reheat this steam after it leaves the high-pressure section of the steam turbine. More specific operating parameters are detailed under each core description.

The reactor and moderator systems and the refueling machine are housed in a cylindrical pressure containment building. All other plant systems and components, including the steam generators, are located outdoors or are housed in conventional buildings appropriately grouped around the reactor building. The main coolant piping is equipped with double shut-off valves where it penetrates the reactor building.

2.3. Thorium Metal Core Designs

2.3.1. Thorium-Optimized Core

2.3.1.1. Core Description

The design of a low-enriched thorium metal nested-cylinder core for the 1000-MWe Thorium HWOCR is described in this section. Design parameters are listed in Table 2-1. The reactor has 324 pressure tubes on an 11.6-inch square pitch, surrounded by D₂O in a 21-foot, 4-inch-ID moderator tank. Each pressure tube has a 5-inch ID and contains five fuel elements stacked axially to form an active fuel length of 18 feet, 9 inches. Coolant flow and refueling are bi-directional. The coolant flows in opposite directions in adjacent pressure tubes and in the same direction as the fuel movement.

The Santowax coolant flow rate through the reactor is 119×10^6 lb/hr at a header-to-header pressure drop of 239 psi. The total system pressure drop is 339 psi. The coolant transports 3019 MW of heat while increasing its temperature from 560 to 708 F. In addition to the heat transported by the coolant, the reactor generates 158 MWt in the moderator and 10 MWt in the shield for a total output of 3187 MWt. This output will produce 1076 MWe at a net plant efficiency of 33.8%.

The fuel element is an assembly of four fuel tubes arranged concentrically around a central moderator can to form annular coolant channels (Figure 2-1). The coolant gap between adjacent tubes is 0.150 inch, the exterior coolant channel is 0.110 inch wide, and the coolant channel between the moderator can and the inner fuel cylinder is a 0.110-inch gap. The can is orificed to restrict bypass flow. The 46.3-inch length of the fuel element includes 45 inches of active fuel in cylinders 1, 2, and 3, and 44.55 inches of active fuel in cylinder 4.

The annular geometry allows optimum thermal conditions for each cylinder and flow channel, so that an average of 9.3 MWt can be removed from each pressure tube without exceeding conservative design limits.

Conversion of Th-232 to U-233 is enhanced by continuous refueling, since the use of poisons to hold down excess reactivity is minimized. With an initial enrichment of ≈ 2.4 wt % U-235, the equilibrium conversion ratio is about 0.86 at 20,000 MWD/T burnup.

A pre-fouling maximum surface temperature of 820 F has been selected for this design to avoid excessive decomposition and fouling of the coolant or hydriding of the cladding. Protuberances on the heated surface of the fuel cylinders (such as local spacers) are avoided to prevent local coolant stagnation. This prevents coolant charring, which could lead to progressive fouling and flow blockage. The rate of hydriding of Zircaloy has been shown to be strongly temperature-dependent. Central fuel temperature is limited to about 1000 F by accelerated irradiation growth rates; however, this temperature is not reached in the fuel because of the surface temperature limit and the good thermal conductivity of the metal fuel.

2.3.1.2. Mechanical Design

The fuel cylinders are made by a coextrusion process in which the Zircaloy cladding (0.025-inch thick) is bonded to the low-enriched thorium metal. The cylinders are extruded in long lengths and then cut to the required length. The fuel is tapered at either end of the cylinder as a result of the extrusion process, and the end cap is extruded as an integral part of the cylinder. Sealing with end rings, with associated problems of machining, brazing, and welding, is eliminated by this process. More importantly, it reduces the axial peaking between elements. The use of the extruded end cap depends upon the success of the development program outlined in section 3.2.

The coextrusion fabrication process has been developed by NMI, SRL, and PNL, using both thorium and uranium metal. The process is adaptable to remote fabrication and recycle of the U-233 produced in the reactor.⁵

Figure 2-1 illustrates the assembly of the fuel element. The four fuel cylinders are supported and pinned (to eliminate

all welding at final assembly) to the base plate, which is a rim attached to a center hub with six radial spokes. A comb effect is incorporated in each spoke to pin and locate the plates for accurate control of the coolant channels. The fuel cylinders are assembled to the comb by line drilling to receive the six 0.188-inch-diameter Zircaloy pins. Since the pins are captured within the fuel element and the pressure tube, no tack welding is needed. The base plate is designed so that the outer rim will accept refueling equipment. The top end cap of the outer cylinder has the same geometry to permit handling at that end. This design is adaptable to an AI-CE refueling scheme. Cylinder 1 has two raised lugs at the top that fit into two mating slots in the lower end cap of the axially adjacent cylinder to prevent rotation of one element with respect to the other.

The flow area in the base plate is greater than that within the fuel element to minimize entrance effects on flow distribution. Six full-length straight spacer ribs on each cylinder will maintain adequate flow channels. The most promising method of attaching the ribs appears to be coextruding a butt rib with the basic cylinder; the remainder of the rib is electron-beam welded to the butt. The spacers are then machined to a finished diameter consistent with assembly requirements. However, NMI believes it may be possible to extrude full-height ribs.

Five elements are stacked axially in a reactor pressure tube to form a fuel column. During its lifetime in the reactor, each fuel element occupies successive positions in the fuel channel; thus the burnup in each fuel element is the same in a given pressure tube at the end of life. Axial loads due to pressure, drag, and weight are imposed on the fuel elements and are taken through the cylinders. Differential axial growth between cylinders will be accounted for by machining the length of each cylinder to agree with its axial growth.

An analysis indicates that vibration of the cylinders is insignificant. The cylinders are short and rigid and could conceivably vibrate as a body, causing wear on the surface of the fuel channel. Although this is not considered to be a problem, since numerous full-length spacers are provided to minimize local contact, further

design analysis and flow tests will be required to determine the nature and magnitude of vibrations, if they exist.

Zircaloy has been selected as the cladding for the metallic fuel, and it is expected to perform satisfactorily if the concentration of hydrides does not exceed an allowable limit. For design purposes, an upper limit of 500 ppm was selected. Several courses of action can extend the design life of Zircaloy: use of improved zirconium alloys, development of barriers, reduced operating temperature, addition of hydride sinks, and increased clad thickness. It is likely that the spacer ribs will provide an effective hydrogen sink.

A first-order estimate of hydride rate was made using the hydride pickup in 300 days for each refueling cycle. Then, assuming the hydride rate is linear with time, the time to reach 500 ppm in five cycles (corresponding to the successive positions in the core) was obtained by calculation using a simple proportion. The following tabulation lists the hot channel surface temperature (including flux tilt factor) of the downstream (hot) end of the fuel cylinder for each position and the hydride pickup in 300 days at that temperature. The total is the hydride pickup in 1500 days. The number of days to reach 500 ppm and the corresponding average burnup are 695 days and 21,800 MWD/T.

<u>Location</u>	<u>Temp, F</u>	<u>Hydrogen, ppm</u>
1	740	145
2	765	185
3	788	242
4	815	330
5	758	175
Total H ₂ (1500 days)		1077

Zircaloy-4 has a better hydride rate than Zircaloy-2 and is more readily available than zirconium-niobium alloys. The slight superiority of the zirconium-niobium hydride behavior (over that of Zircaloy-4) does not warrant the extra cost at this time.

Zircaloy and thorium metals are compatible and do not form objectionable intermetallic compounds at the fuel-clad

interface. Aluminum cladding (metal or SAP) is not suitable for the metal fuel, because aluminum forms brittle intermetallic compounds, and because thorium diffuses into aluminum. Aluminum metal is also unsuitable because of its low melting point. Stainless steel cladding was not considered because of its high neutron cross section.

The thermal expansion coefficient of the Zircaloy cladding is less than that of thorium. In operation, this difference will cause thermal strain within the cylinder, but this is not expected to be a problem because thermal cycling will be minimal with on-line refueling. Analysis and testing will be required, however, to establish allowable design limits (see section 3.2).

Several fuels have been developed and tested for the nested-cylinder configuration for various reactor designs and concepts. Powdered ceramics have been tried in thin-walled clad cylinders, uranium and thorium alloys have been co-extruded with Zircaloy cladding, and attempts have been made to cast carbide rings. For this reference design, the use of thin-walled nonbonded cladding has been discarded because the thin-walled cladding would not withstand the pressure buildup caused by fission gas release from oxides, carbides, etc. Metallic thorium fuel is more stable than uranium, is compatible with Zircaloy, and can be bonded with the Zircaloy cladding. Therefore, thorium alloys are more desirable for the nested-cylinder configuration, and better developed, than are oxides and carbides.

The irradiation behavior of thorium fuels is reviewed in detail in section 3.2. In the metal fuels, fission products are not released, as they are in ceramic fuels, to cause fission gas pressure in the cladding; instead they are retained within the metal, causing changes in the density and dimensions of the fuel.

The effect of fuel growth on the fuel cylinder geometry may be complex, making predictions of dimensional changes difficult. Thorium cylinders tested at Savannah River experienced some growth of the outer cladding and very little change in the inner cladding.⁶ Uranium cylinders, however, experienced buckling of the inner cladding. If the fuel cylinder were allowed to grow unrestrained by the cladding, the cylinder would expand in the same manner as a tube which is heated and expands thermally. In this case, both OD and ID would increase

proportionately. Cladding restraint, however, may cause the higher-temperature fuel to flow along the path of least resistance. It is expected that the initial effect of growth will be to stretch the outer cladding, an effect experienced with thorium cylinders at SRL, with relatively small changes in diameter. Very large growth or swelling would probably cause distortion, as experienced with uranium cylinders. The dimensional changes are expected to vary, depending on cladding thickness, cylinder geometry, burnup, and fuel temperature. No generalizations can be made, therefore, except that available irradiation data for thorium cylinders show promise of acceptable performance.

Fuel growth was calculated based on a peak burnup of 38,000 MWD/T, a growth rate of 2% volume change per atomic % burnup, and the assumption that the growth will stretch the outer cladding without changing the inner cladding. The results of this calculation show a change in flow channel thickness of 7 mils at the time the fuel assembly is discharged from the reactor. The effect of growth on flow channel thickness is included in the thermal and hydraulic calculations.

Fabrication tolerances were also studied. The dimensional variations of thin-walled tubing increase when the wall-to-diameter ratio decreases. For the larger-diameter cylinders, the wall thickness was limited to 0.100 inch or larger. The 5-inch-diameter cylinders are larger than any experimental cylinders made to date; consultation with personnel from NMI, SRL, and PNL indicates that extrusion of cylinders of this size appears to be feasible. With respect to fabrication tolerances, three general areas of variations in fuel dimensions are of primary concern: fuel, cladding, and coolant channels. The following values were used in design analyses:

Fuel Variations

Core thickness/fuel tube

Local	Nominal $\pm$ 5%
Avg thickness/fuel cylinder length	Nominal $\pm$ 3%
Fissile fuel uniformity/fuel tube, local	Nominal $\pm$ 5%

The variation affecting heat transfer to the coolant is significantly reduced by integration over the length of five elements in a fuel channel. For design purposes, a value of $\pm 5\%$ in the average fuel variation in a fuel channel was used.

Cladding Variations

Cladding thickness/fuel tube

Local	0.025 ± 0.005
Avg thickness/fuel cylinder length	0.025 ± 0.003

Coolant Channel Variations (mechanical tolerances only)

Tolerance in nested cylinders with straight ribs was studied. On the basis of discussions with NMI, SRL, and PNL personnel, a straightness of 0.010 in./ft (for the basic cylinder) was used in arriving at these tolerances. Of all the tolerances considered, straightness has the greatest effect on channel variation. The straightness achieved in practice is indicated below:

PNL	0.015 in./ft
SRL - as extruded	0.938 in./12 ft
SRL - after straightening (enriched U-Th)	0.015 in./ft
SRL - after straightening (natural U-Th)	0.010 in./ft

Figure 2-2 shows the bow condition. The following factors were assumed in the tolerance analysis:

1. ID clad ± 0.005 in.; OD clad ± 0.005 in.
2. OD ribs ± 0.003 in.
3. Coextruded cylinders resized and straightened to achieve required diameter and straightness tolerances.
4. Bow 0.010 in./ft.
5. Six straight ribs electron-beam welded to co-extruded stubs, cylinder machined on OD of ribs to ± 0.003 in. (when mounted) to eliminate bow on the ribs.
6. Assembly clearance between cylinders 0.007 in. on the diameter.

A typical calculation for cylinder 1 - 2 produced the following results (all units in inches):

ID of cyl 2	2.970
	(-) <u>0.034</u>
OD of ribs, cyl 1	2.936 ± 0.003
Basic rib height =	0.133 ± 0.006

Side "A" Channel (100% tolerance stack-up)

Basic rib height	0.133 ± 0.006
Max channel	0.158
Min channel	0.108
Avg channel =	0.133
Variation =	± 0.025

Side "A" Channel (80% tolerance stack-up)

Variation =	$0.80 \times 0.025 = \pm 0.020$
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The ± 0.020 in. figure is the maximum variation in one channel for one element at the 80% tolerance stack-up. This considers the maximum adverse bow position for both cylinders. In traversing the length of the fuel channel, the average size variation of the channel is much less than the local variations. A variation of ± 0.010 inch in the average channel size is allowed for fabrication tolerances in thermal and hydraulic calculations.

2.3.1.3. Nuclear Design

Nuclear calculations were made to obtain power distributions and data for fuel cycle cost calculations. Continuous refueling is simulated by doubling the lifetime of a batch-loaded core (as described in BAW-393-2¹). Since this technique approximates the approach to equilibrium rather poorly, a more accurate but time-consuming backup calculation was run to verify the equilibrium cycle obtained from the double-batch recycle. In this calculation, the core is loaded initially to run as a true batch core for about 8000 MWD/T, at which point "continuous" fueling is initiated by refueling 1/5 of the core at a time, at intervals so that discharge burnup is the same as in the double-batch case (20,000 MWD/T). Fuel added to the reactor each time is sufficient to keep the reactor k_{eff} at an average of 1.000 over the next time interval. In principle, this calculation is very similar to the TONG code used at ORNL except that, because of automation, TONG can refuel a much smaller fraction of the core at a time. The time-consuming nature of the calculation prevented recycling completely to equilibrium, but results indicate that the equilibrium cycle computed with the double-batch lifetime recycle method is in good agreement with the more refined continuous refueling model. The results of the nuclear calculations are

given in Table 2-1, part IV. Part IV contains mass balance data from the U-235 double-batch recycle calculation and results through the third complete recycle of all fuel from the continuous refueling case. All masses are reported on a basis of one pressure tube. Each batch is representative of the entire reactor in the double-batch case and of 1/5 of the reactor in the continuous refueling case. Protactinium decay of 180 days was allowed, along with 1.5% total losses during reprocessing and refabrication. A minimum holdup of one year was assumed before recycle fuel could re-enter the reactor. Subsequent information indicates that this can be reduced to less than 300 days, which would reduce the out-of-pile inventory considerably.

Calculations were also run for plutonium as an alternate startup fuel. Results of the double-batch recycle calculation based on recycle of bred plutonium fuel are included in part IV of Table 2-1. The plutonium mixture used for startup and feed is that computed by ORNL as being discharged from the Uranium HWOCR.

Pu-239	52.8%
Pu-240	28.8%
Pu-241	12.7%
Pu-242	5.7%

U-233 idealized-cycle calculations were done for the 4-cylinder design for comparison with previous analyses by the same method. The idealized cycle has a representative fuel loading of 94.5% U-233 and 5.5% U-234. This particular fuel combination was chosen arbitrarily for parameter studies and design evaluation on a consistent basis without requiring recycle calculations. Results of the idealized-cycle calculations are included in Table 2-1, part IV, along with the neutron balances for the idealized-cycle, U-235-recycle, and plutonium-recycle cases.

Reactor power distribution is broken down into four components for heat transfer calculations: gross radial, gross axial, local radial, and local axial. All but the local radial are independent of the nested-cylinder radii. The local radial is an outgrowth of the nuclear-thermal iterations used to size the cylinders for maximum performance. A discussion of each component follows.

Gross Axial $P/\bar{P} = 1.30$

The gross axial peaking factor was computed with a three-dimensional synthesis calculation which accounts for counter-current fuel movement. This method, derived completely in BAW-393-2,¹ produced a peaking factor of 1.33 for a clustered-pin design with a conversion ratio of 1.016. Since conversion ratio directly affects the distribution of fissionable material along the fuel channel, it was conceivable that the axial peaking would be tied strongly to the conversion ratio. A second synthesis case, using equilibrium cycle data for a similar design with a 0.72 conversion ratio, showed that power peaking is not strongly dependent on the conversion ratio. The axial power distributions (Figure 2-3) deviate from a chopped cosine, but the magnitude of the peak-to-average power ratio is relatively unaffected. This comparison indicates that counter-current fuel movement has a stabilizing effect on the gross axial power distribution. Since the 4-cylinder thorium metal design has a conversion ratio of 0.86, the gross axial peaking factor will probably fall between the two curves shown in Figure 2-3. A value of 1.30 was selected.

Gross Radial $P/\bar{P} = 1.17$

The core will be zone-loaded in three radial zones to achieve power flattening. A small differential in enrichment will suffice to produce the large degree of power flattening desired. Calculations made with the U-233 idealized cycle indicate that a difference of less than 10% between maximum and minimum enrichment will provide a peak of less than 1.17. This applies equally to the nested-cylinder and clustered-pin cores. Exact enrichments have not been computed for the final design, but the increased reactor leakage due to zone loading has been accounted for in the lifetime calculations. Figure 2-4 shows the predicted radial power shape.

Local Axial $P/\bar{P} = 1.10$

Local axial peaking arises because the fuel-bearing section is not continuous over the entire length of the element. Peaking factors as high as 1.7 to 1.8 occur on the innermost cylinders. The simplest way to combat this is to taper the ends of the cylinders in such a way that the heat flux resulting from the peaking is not excessive.

Each cylinder will be tapered differently in proportion to its peaking. Proper use of the taper should hold the heat flux peaking factor below 1.10. Figure 2-5 shows the actual peaking from PDQ calculations before the taper is introduced.

Local Radial

Local radial peaking is unique to each design and each fuel material. It results from nuclear-thermal iterations in which the cylinder radii are variables. Optimum performance is obtained if the power distribution within the element is constant during life, since each ring can be sized for the maximum peaking.

Initial fueling with U-233 (the idealized cycle) does indeed produce this behavior—a near-constant power shape with life; however, U-233 is unavailable for fueling, and the use of U-235 for startup produces a total swing in peaking (for any given ring) of about 12% between cycle 1 and equilibrium. The cylinders can be sized, as a compromise between the two distributions, with a total reduction in element power of only about 6%. Local zone-loading may further alleviate this.

Plutonium presents a greater problem. Initial peaking is 25% higher than for U-235, and the total swing over life is about 48%. Methods to counter this change have not been developed, and at this stage, it appears that U-235 will perform better in the nested-cylinder design.

Local radial peaking factors are tabulated below. The cylinder configuration used in this comparison of peaking factors was optimized on the basis of idealized-cycle calculations. Nuclear-thermal iterations could size the cylinders to get about the same power at time zero in each case without appreciable changes in other characteristics. The change in peaking as a function of time is the important characteristic to be noted.

Local Radial Peaking ($P/\bar{P}$)

	Fuel					
	U-235		Plutonium		U-233 (idealized)	
	Cycle 1	Equilibrium	Cycle 1	Equilibrium	BOL	EOL
Cyl 1	0.58	0.65	0.47	0.69	0.65	0.69
Cyl 2	0.71	0.77	0.54	0.74	0.77	0.79
Cyl 3	1.09	1.06	1.05	1.01	1.09	1.07
Cyl 4	1.80	1.61	2.28	1.65	1.64	1.58

2.3.1.4. Thermal Design

The hottest pressure tube in the core has a nominal power output to the coolant of 10.9 MWt, and the average fission power per pressure tube is 9.8 MWt. The temperature of the coolant at the outlet of the hot pressure tube is 708 F. The distribution of coolant flow to the 321 pressure tubes is controlled by orifices sized according to the radial power distribution to provide a mixed-mean reactor outlet temperature of 708 F.

A computer program, TANC (described briefly in BAW-393-2 and detailed in BAW-290), was used to evaluate the performance of the nested-cylinder core.^{1,7} This code determines the heat split in each cylinder of the element, calculated maximum fuel and surface temperatures, and the coolant temperature rise in each channel.

The thorium metal nested-cylinder thermal and hydraulic analysis is based on the hot channel concept to assure adequate cooling of the fuel elements under the most adverse conditions. The hot channel is defined as the channel with the poorest heat transfer capability, the largest effective stack-up of tolerances, and the highest power generation. In determining hot channel factors, specification of too-optimistic conditions indicates performance beyond the true reactor capability. On the other hand, consideration of the most-adverse local conditions is also misleading, since channel size averaged over the length of five separate elements stacked axially in the pressure tube is the effective parameter that retards coolant flow and increases coolant temperature rise. These and other similar factors were considered in specifying the

variations from the nominal conditions given in section 2.3.1.2 and used in the hot channel analysis.

Since the input data to the TANC code allows dimensional and power variations, the hot channel geometry and heat generation were "mocked up" to simulate actual operating conditions more accurately than by use of hot channel factors. The power output of the hottest fuel cylinder was increased by an additional 5%, and the thickness of the adjacent coolant channels was decreased to reflect the most-adverse manufacturing tolerances and fuel growth. In essence, this method imposes a coolant temperature rise hot channel factor of 1.16 on the channel enclosed by the two highest-power cylinders. Figures 2-6 and 2-7 show the nominal and most-severe operating conditions. The burnout ratio as a function of length (Figure 2-8) shows that the minimum burnout ratio is about 2.5. Other significant parameters are included in Table 2-1, part III.

2.3.2. AI-CE Design With Thorium Metal Fuel Elements

2.3.2.1. Core Description

This section contains the results of a preliminary effort to define the characteristics of a thorium metal nested-cylinder element for operation in a 1000-MWe Uranium HWOCR. Design parameters are listed in Table 2-2.

Two difficulties are immediately encountered: the 1000-MWe uranium plant is not defined, and the power capabilities of the UC pin and thorium metal nested-cylinder elements are different. Since current design information is unavailable for the 1000-MWe uranium plant, it was assumed that, basically, the core internals correspond to those of the 500-MWe Uranium Demonstration Plant described in AI-CE-MEMO-25;⁴ the nominal 1000-MWe (1076-MWe actual) plant was defined by extrapolation, keeping the maximum power per pressure tube and the net plant efficiency constant. The larger core consists of 548 pressure tubes or fuel channels on a square pitch of 11.0 inches. Each pressure tube assembly has an ID of 4.030 inches and contains five fuel elements stacked axially to form an active fuel length of 17 ft, 9.3 inches. The Zircaloy outer tube forms a CO₂ insulating annulus in combination with the SAP inner tube and is surrounded by the D₂O moderator, which is contained in a 26 ft, 3-inch-diameter moderator tank.

A one-pass, bi-directional coolant system is utilized with coolant flow in the direction of fuel movement. Bi-directional on-line refueling is accomplished by two refueling machines located at the top and bottom of the core with fuel movement in opposite directions in adjacent pressure tubes.

As mentioned earlier (and shown in Table 2-2) the power capability of the nested-cylinder element is about 30 to 35% higher than that of the UC pin-type element described in AI-CE-MEMO-25.⁴ This high power capability is an inherent characteristic of the nested-cylinder geometry (see section 2.3.3). While favorable from an economic standpoint, the higher power capability leads to some difficulty in adapting the design for operation in the 1000-MWe Uranium HWOCR. For example, with a fixed geometry either the core must operate at its potential and produce a significantly higher power output, which is incompatible with the secondary side equipment, or it must operate at a specific power far below its capability, which penalizes the economic performance. Naturally, if excess capacity could be assumed in the secondary side equipment, the power costs would be reduced by about 30 to 35% because of the extra power produced, but this assumption would be unrealistically favorable to the thorium-fueled HWOCR. Rather than adapt either of these assumptions, the economic performance of the nested cylinder core was based on operation of the element at its predicted capability in the number of 4.03-inch-ID pressure tubes required to achieve 1076-MWe, to illustrate its potential more accurately. The excess pressure tubes in the AI-CE plant would not be utilized for power production.

As shown in Table 2-2, 414 pressure tubes or fuel channel assemblies are required. At the average fission power of 7.76 MWt per pressure tube, the reactor produces 3209 MWt, of which 3039 MWt are transferred to the coolant and 170 MWt are generated in the moderator or other structures. Santowax coolant enters the core at 560 F, flowing at the rate of 119×10^6 lb/hr, and exits at 711 F. The total system pressure drop is 400 psi with 300 psi differential from header to header across the core. Since pressure differentials, inlet and outlet temperatures, etc., do not match those of the 500-MWe Uranium Demonstration Plant, relatively minor adjustments to coolant channel sizes would be required if the 500-MWe plant characteristics are main-

tained in the revised 1000-MWe design. These adjustments, together with final thermal-nuclear iterations to produce a fully optimized element, could lead to a four-cylinder element similar to that for the 5-inch-ID pressure tube.

2.3.2.2. Mechanical Design

Fuel Element Description

The basic cross section of the element is shown in Figure 2-9. It is an assembly of five fuel tubes arranged concentrically to form annular coolant channels with a flow blocker in the center of the element. The coolant gap between adjacent fuel cylinders is 0.125 inch, the exterior coolant channel is 0.100 inch, and the channel between the center flow blocker and cylinder 1 is 0.110 inch. The fuel thicknesses are 0.284, 0.240, 0.175, 0.126, and 0.090 inch for cylinders 1, 2, 3, 4, and 5, respectively. The fuel element is 43.96 inches long, of which 42.66 inches is active fuel in cylinders 1, 2, 3, and 4, and 42.21 inches is active fuel in cylinder 5.

Metallic thorium alloyed with about 2.4 wt % uranium is used in the fuel tubes. The fuel cylinders are made by the process used in the four-cylinder design. The general description of the four-cylinder center moderator design (in section 2.3.1.2) applies to this element except as noted previously.

2.3.2.3. Nuclear Design

Physics calculations for the AI-CE pressure tube were confined to the usual nuclear-thermal iterations to size the cylinders, followed by a running of U-233 idealized-cycle data for cost comparison with the thorium-optimized core. Nuclear characteristics are nearly identical to those of the thorium-optimized design, i.e., conversion ratio, specific power, and enrichment.

Table 2-2 summarizes the U-233 idealized-cycle data for the nested-cylinder design in the AI-CE pressure tube. Local radial peaking data are also included in the table. The other three components of the power distribution (gross axial, gross radial, and local axial) are the same as those described for the thorium-optimized 1000-MWe HWO CR.

2.3.2.4. Thermal and Hydraulic Characteristics

The hottest pressure tube in the core has a nominal power output to the coolant of 8.6 MWt, and the average fission power per pressure tube is 7.76 MWt. The temperature of the coolant at the outlet of the hot pressure tube is 711 F. The distribution of coolant flow to the 414 pressure tubes is controlled by orifices sized according to the radial power distribution to provide a mixed-mean reactor outlet temperature of 711 F.

The calculation methods and computer programs used to determine the thermal and hydraulic characteristics are the same as those used on the four-cylinder element. Figures 2-10 and 2-11 show the nominal and most-severe operating conditions; Figure 2-12 shows that the minimum burnout ratio is about 2.5. Other significant parameters are listed in Table 2-2.

2.3.3. Evaluation of Thorium Metal Core Design

The design optimization studies have shown consistently that the thorium metal nested-cylinder cores have an economic advantage over other types. This advantage is directly related to the predicted performance capability, which depends partly on the design criteria or technical limits imposed on the design and partly on the inherent characteristics involved. Since absolute values on allowable operating conditions or limits are unknown in many instances, engineering judgment must be used. Thus in comparative evaluations, all criteria and assumptions utilized in each design must be examined carefully to avoid undue optimism or pessimism in establishing the performance criteria, and the technical uncertainties must be minimized as much as possible.

The inherently favorable characteristics of the thorium metal cores are attributed more to their adaptability to annular geometry than to the nuclear characteristics of the fuel material. A high-performance, relatively high-burnup, nested-cylinder fuel element that can be fabricated at relatively low cost is more feasible with thorium metal than with oxide or carbide. The relatively simple coextrusion process which bonds the fuel and cladding, together with the high conductivity of the fuel material, results in low fuel temperatures. The low fuel temperature, in turn, improves the irradiation stability and minimizes swelling of the

fuel material, so that relatively high burnups should be achievable. The low fuel temperature and the fuel-clad bond therefore preclude the problem of internal fission gas pressure that is likely to limit the performance of oxide or carbide fuels in low-pressure reactors.

The nested-cylinder geometry that evolves from these considerations is ideally suited to a pressure tube reactor. The circular configuration prevents excess bypass flow, which can present a design problem when clustered-pin elements are placed in the round pressure tube. The elimination of excess flow reduces the amount of coolant (which acts as a poison) in the core and increases the reactor outlet temperature, which in turn improves plant thermal efficiency. The coolant channels between adjacent fuel cylinders can be varied to maintain coolant outlet temperature from all channels at the maximum value, regardless of the radial power variation across the element. The fuel cylinder thicknesses can also be varied to adjust the radial power distribution, which permits the designer to optimize for the maximum allowable surface heat flux on all cylinders and to reduce the amount of surface or cladding material in the core. In other words, the thermal and hydraulic design flexibility of the nested-cylinder geometry permits each cylinder to be designed for maximum power generation, maximum surface heat flux, and maximum cladding temperature. Concurrently, each channel can be designed for maximum outlet temperature at a minimum core pressure drop. These characteristics produce a high power per pressure tube, high reactor outlet temperature, and minimum reactor flow rate to reduce plant capital and operating costs.

The flexibility of the nested cylinder design enhances one of the basic advantages of organic coolants—high coolant temperatures at low operating pressure. Conversely, the use of organic coolant enables the use of thorium metal, since the metal is not highly reactive with the organic, as it is in the case with water.

The characteristics of the nested-cylinder fuel element favor low capital costs in conjunction with the potential for low fuel cycle costs. A successful fabrication development program should result in low manufacturing costs because of the small number of massive cylinders per fuel element. Similarly, the fabrication component of the fuel cycle cost can be reduced by achieving a potentially high burnup, which prorates

the fabrication cost over a larger amount of power and income. The thorium metal has a high density which, along with the low clad-to-fuel and coolant-to-fuel ratios, favors a high conversion ratio and a low fuel burnup expense. The optimum use of surface area and the higher allowable heat fluxes (because of favorable hydraulic characteristics) contribute to an increase in specific power and a reduction in fuel inventory cost. Finally, the higher plant efficiency attainable with the high reactor outlet temperatures and the adaptability of the nested-cylinder element to larger-diameter pressure tubes lead to a reduction in the number of pressure tubes, the amount of D₂O required, and the plant capital cost.

The nested-cylinder fuel element, therefore, does have inherent characteristics which, in theory, give it performance and economic advantages over alternate designs. The extent to which these advantages can be realized in practice depends on the compatibility of the element with the HWOCR environment, the irradiation and operational stability, and the validity of the design assumptions and cost estimates. Since data are not available at projected operating conditions, extrapolations involving judgment and estimates are unavoidable, and the associated uncertainties should be examined carefully.

One area of concern is the dimensional variations of the fuel, cladding, and coolant channels caused by fabrication tolerances, fuel swelling, and operational distortions. Information on all these aspects is available from reports on development programs at PNL, NMI, and SRL (discussed in section 3.2), but irradiation tests were limited to small-diameter, single-cylinder elements. In addition, projected fuel burnup and clad surface temperatures are greater than the values associated with test conditions; however, the maximum fuel temperature is below the corresponding values in test elements. Although extending the results to large, closely-spaced tubes involves some extrapolation, the results from these programs are favorable in all areas of concern. For example, the PNL experiment at about 13,000 MWD/T burnup indicates a swelling rate near 1%/atomic % burnup, whereas 2% was used as a design basis.

Assuming that the projected burnup can be achieved without affecting the structural integrity of the element, the primary effects

of fuel swelling or operational distortions are reflected in the thermal-hydraulic characteristics. Higher core pressure differentials will be required to maintain the specified cladding surface temperatures on the cylinder adjacent to a deviated channel. Consequently, flow rates in nominal pressure tubes are higher, and the coolant outlet temperatures are lower. Both changes affect the net plant efficiency adversely and thereby alter the economic performance. Other variables that play a significant role in determining power cost, such as specific power, would not be altered significantly by reasonable changes in channel dimensions.

The thermal-hydraulic analyses of all nested-cylinder cores have been based on an average coolant channel variation of ± 13 mils. This variation is a significant percentage of the nominal coolant channel. Allowance for this change reduces the coolant temperature rise about 16%, and to some extent negates the potentially superior thermal performance of the nested-cylinder geometry.

Development and test results will be required to further define the fabrication tolerances that can be held at reasonable cost, as well as the changes in fuel element dimensions caused by irradiation. Assessment of possible adverse results indicates that the power cost increases about 0.003 mills per 0.001 inch change from nominal channel dimensions. Correspondingly, an improvement in dimensional control could improve the performance.

Another uncertainty is the hydriding of the cladding. The practicality of successfully operating Zircaloy-4-clad tubes in organic coolant for the two-year period required to achieve the design burnup has not been clearly demonstrated. However, the performance of zirconium alloys in organic coolants has been investigated by AECL since early 1961, and the basic compatibility of the materials has been established. Zirconium-clad pins are currently in operation in the WR-1.

The economic effect of excessive hydride buildup would probably be reflected in reduced burnup limits, which would increase power cost at 0.015 mills/kwhr per 1000 MWD/T reduction. Naturally, if burnups higher than the design value can be achieved, costs will be reduced by a similar amount.

Other factors that could alter the predicted economic performance significantly are involved with fabrication development. In

order to reduce the heat output in these high-axial-peaking-factor zones, a method must be developed for tapering the fuel to controlled dimensions at each end of the fuel element. While this should be successfully accomplished during the fabrication development, it is essential in order to maintain the design specific power. Another requirement that must be met to maintain the high power capability of the element is successful coextrusion of the relatively large-diameter cylinder with a 0.100-inch fuel thickness. Like the end taper situation, early fabrication-development effort should demonstrate achievement of this objective and could show that thinner cylinders are feasible. This development would permit an increase in the economic optimum specific power.

Some difficulty is encountered in adapting the thorium nested-cylinder design for operation in the UC-fueled HWOCR at its optimum economic performance because of the higher power capability of the cylinder geometry, as previously noted. However, some difficulty is inevitable when designs utilizing alternate fuel materials are adapted to a plant designed for fuels with significantly different properties and characteristics. The alternate designs usually suffer in capability, because physical characteristics of the plant dictate nonoptimum core designs. The degree to which each design is penalized, of course, depends on which are the controlling properties of the fuel material or geometry and on the flexibility inherent in the basic design. The design parameters and flexibility of the plant also are influential in determining the success of the efforts. While considerably more effort will be required to establish the best method of adapting the design to the AI-CE plant, all evidence indicates that it could be accomplished successfully once the plant design is more firmly established.

Development of the thorium metal nested-cylinder element would offer backup technology for the primary HWOCR program in conjunction with development of the thorium fuel cycle. Since the low-ductility SAP cladding has not been fully developed for operation at currently specified design conditions, adverse results could downgrade the projected performance of the HWOCR. Thus, establishment of the Zircaloy as a successful cladding, which is an inherent part of the nested-cylinder development, provides a valuable backup to the primary program. Developing the thorium metal fuel element would also complement the effort

on advanced and dual-purpose HWO CR designs since annular geometries are being considered for these applications.

In summary, the thorium metal nested-cylinder fuel element has a number of inherently favorable characteristics, and development of this element would contribute to the primary HWO CR program by providing alternate materials and geometries as backup. At the same time, there are unfavorable aspects that could minimize the advantages. The performance analysis has allowed for these aspects, based on judgment and extrapolation of existing data. The adequacy of these allowances naturally depends upon the success of the development effort, and further definition of the achievable performance must await the accumulation of fabrication and operational data. While the current element differs somewhat from the design reviewed by ORNL, the following conclusion from the draft HWO CR Evaluation Report still summarizes the situation adequately:⁸

"The thorium metal nested tube fuel element appears to be well designed, and it should be operable at the heat ratings and temperature limits proposed. It represents a reasonable extrapolation of present technology, but additional testing is required to demonstrate that projected life and burnup can be achieved under reactor operating conditions. It must be shown that fuel element distortions resulting from thermal gradients or fuel swelling do not adversely affect operation of large, closely spaced nested tubes, and also that the many factors affecting hydrogen pickup and hydride embrittlement can be controlled to ensure satisfactory life for fuel elements of this geometry."

2.4. Thorium Oxide Core Designs

2.4.1. Thorium-Optimized Core

2.4.1.1. Core Description

The thorium oxide clustered-pin core for the 1000-MWe Thorium HWO CR is described in this section. Design parameters are listed in Table 2-3.

The reactor complex arrangement for the clustered-pin design is similar to that of the nested-cylinder core. Basically, the core consists of 758 pressure tubes or fuel channels on a square pitch

of 9.5 inches, surrounded by D₂O in a 26-foot, 7-inch-ID moderator tank. Each pressure tube has an ID of 4.030 inches and contains five fuel elements (equivalent active fuel length per element is 42.66 inches) stacked axially to form a total active fuel length of 17 feet, 9.4 inches. Each fuel element has 37 pins arranged on a triangular pitch. The pins are composed of SAP 895 cladding filled with Vipac ThO₂-UO₂ to 88% theoretical density.

A one-pass coolant system is utilized as in the nested-cylinder core. Coolant flow and refueling are bi-directional. The coolant flows in opposing directions in adjacent pressure tubes and flow is in the same direction as the fuel movement. Santowax coolant flows through the core at the rate of 107.2×10^6 lb/hr at a total core pressure differential of 146 psi, including header and shield plug losses. The pressure drop across the active core is 92 psi; the total system pressure drop is 246 psi. The coolant transports 2936 MW of heat while increasing its temperature from 560 to 721.4 F. In addition to the heat transport in the coolant, the reactor generates 157 MWt in the moderator and 10 MWt in the shield for a total output of 3100 MWt. This output will produce 1076 MWe at a net plant efficiency of 34.7%.

2.4.1.2. Mechanical Design

Vipac sol-gel ThO₂-UO₂ fuel is well suited for remote fabrication to permit recycle of the U-233 produced in the reactor. Very little work has been done with the combination of SAP cladding and ThO₂-UO₂ fuel, but the nature of the vibratory-compacted fuel should preclude any incompatibility between the cladding and fuel that might be caused by thermal and irradiation growth.

SAP 895 was selected as the cladding material on the basis of the AI-CE work. Final establishment of the cladding thickness will be based on the cladding being free-standing and on its ability to withstand the internal and external pressure loading and the hydraulic and gravity loads that act on the fuel pin. The creep and stress-rupture properties of SAP 895 at operating conditions will govern the ability of the cladding to withstand these loads. Based on preliminary work and engineering estimates, the 0.022 and 0.017-inch-thick cladding will meet the requirements; further evaluation is necessary, however.

The fuel element has 37 fuel pins arranged on a triangular pitch in an array that has a circular outer boundary to fit inside a 4.030-inch-ID pressure tube (Figure 2-13). The 31 large fuel pins have a fin diameter of 0.631 inch, a 0.515-inch clad OD, and a 0.022-inch wall thickness. Twelve fins, 0.058 inch in height by 0.030 inch thick, spiral on the outside of the clad with a 90°/ft helix pitch. The fins extend over the active fuel length. The six small pins have a fin diameter of 0.408 inch, a 0.332-inch clad OD, and a wall thickness of 0.017 inch. Six spiral fins on the outside of the clad extend over the active fuel length. Two of the fins are 0.038 inch high by 0.030 inch thick, and four are 0.102 inch high by 0.030 inch thick. A machined cut on the two sides of the small pins gives the geometry shown in Figure 2-13. The overall element length is 44.82 inches, including an equivalent active fuel length of 42.66 inches. Void space is 1.300 inch in the large fuel pin and 0.430 inch in the small pins.

The 37 fuel pins are located and retained by two end grids. The Zircaloy end grids are circular ferrules, electron-beam welded or brazed in the arrays shown in Figure 2-13. The best method for joining the grid ferrules appears to be electron-beam welding, but brazing will also be investigated to ensure the optimum compromise between structural integrity and economics. The ferrule-type grid has less metal in the core and greater flow area than the plate-type end fitting. It is presented as an alternate to the plate-type end fitting used in the AI-CE design.

The diameter of the ferrules has been increased by 0.002 inch (compared to the fin diameter on the fuel pins) to accommodate differential thermal growth and adverse tolerance buildup (which would affect assembly of the element) and to reduce the clearance between the element and the pressure tube to cut down eccentric bypass flow. The diametral clearance at the end grids between the fuel element and the pressure tube amounts to 0.047 inch. At the midpoint of the element, the diametral clearance is 0.060 inch.

The 12 large-diameter fuel rods on the periphery of the element are fixed to each end grid by disc ferrules and Zircaloy cap screws. The 6 small-diameter fuel rods are retained by specially-shaped ferrules in each end grid. The 19 inner fuel rods are positively

positioned by dimple indentations on the inner ferrules. A projection on the bottom end cap of each fuel pin fits into a slot in each ferrule of the bottom grid to prevent rotation of the pin. A projection on the top end cap of each of the 19 inner fuel rods stops rod movement in that direction. The end caps are attached to the cladding by magnetic-force-resistance welding.

Five fuel elements are stacked vertically in a pressure tube. The axial pressure and weight loadings on each element are carried through column action by the 12 large and the 6 small pins on the periphery. AI-CE tests on a similar element indicate that pins in this diameter range do not need mid-element straps to prevent excessive vibration. Two spikes, which are an extension of the ferrules in the top grid of an element, mate with specially-shaped ferrules in the bottom grid of the axially adjacent element to index one element relative to another. The element is handled at the outer rim by lifting pawls on the AI-CE refueling machine. Some modifications will probably be required to ensure proper mating with the AI-CE equipment.

2.4.1.3. Nuclear Design

Table 2-3 summarizes the results of physics calculations for the economically optimum thorium oxide clustered-pin design. The physics methods are described in BAW-393-4.² Results given in Table 2-3 include data for the U-233 idealized cycle, U-235 startup and recycle, and plutonium startup and recycle. (The idealized cycle is defined by an arbitrary fuel loading which is 94.5 wt % U-233 and 5.5 wt % U-234.) Recycle analyses were made using the following ground rules: 180-day decay time for protactinium, 1.5% reprocessing losses, and a makeup mass sufficient to keep the same total mass of metal for the start of each cycle. The U-235 recycle assumes uranium that is 93 wt % U-235 for both startup and makeup; the plutonium recycle assumes a plutonium mixture that is 52.8 wt % Pu-239, 28.8 wt % Pu-240, 12.7 wt % Pu-241, and 5.7 wt % Pu-242 (this being the calculated output from a uranium HWOCR plant) for both startup and makeup. Because of uncertainties in spectrum and cross sections, the plutonium recycle data are not as reliable as the U-235 recycle data.

Power peaking factors are included in Table 2-3; THERMOS-corrected, local radial power peaking factors are given for

the idealized cycle at time zero. For U-235 startup, these peaking factors will be about 5 to 7% higher (see Table 2-5); therefore, minor zone-loading would be required for the startup elements to achieve the idealized-cycle values. The results of RZ PDQ calculations of the local axial power peaking are shown in Figure 2-14. The increase in the axial peaking factor toward the center of the element is a function of the relative contribution of thermal neutrons from the D_2O moderator and from the non-fuel region between the axially adjacent fuel elements. Since the resonance energy flux is flat, the slowing-down source from the non-fuel region does not diminish toward the center of the element; however, the thermal neutron supply from the D_2O is sharply attenuated toward the center. Therefore, at the ends of the element, a higher fraction of the thermal neutrons supplied to the innermost fuel comes from the non-fuel region. The logical consequence is a more pronounced gradient near the ends of the element and higher axial peaking toward the center. Although the axial peaking factor at certain points in the center may be as high as 1.6, the total heat generation rate (including the local radial peaking factor) is less than the total in the outer fuel. Therefore, the high value of the axial peaking factor at the center of the element does not significantly affect the thermal performance of the clustered-pin design. Design changes can be made to reduce the local axial peaking factor. Reduced or non-enriched pellets at the ends of the fuel rods may be used in the clustered-pin design. This is a common approach that has been successful in other reactors. A local axial peaking factor of 1.17 was conservatively selected for hot channel (outer fuel region) calculations.

Gross axial and radial power peaking factors of 1.3 and 1.17 were used in design analyses. These values are based on the analyses reported previously for the nested-cylinder core (section 2.3.1.3). Although detailed power peaking calculations remain to be done for the clustered-pin design, large variations in gross power shapes between the various cores are not anticipated. The clustered-pin design possesses some nuclear flexibility in that some control over both the axial and radial power peaking can be achieved through the use of such techniques as zone-loading, depleted fuel, etc.

The validity of the standard lifetime physics model for determining the equilibrium cycle of a continuously refueled and recycled core has been tested with a more sophisticated mockup of continuous refueling; this is discussed in section 2.3.1.3.

2.4.1.4. Thermal Design

Thermal and hydraulic parameters of the thorium oxide clustered-pin core are listed in Table 2-3. The hottest pressure tube in the core normally transfers 4.53 MWt to the coolant. Including allowance for a 1.05 flux tilt, the maximum power capability is increased to 4.75 MWt. The average pressure tube fission power is 4.09 MWt. The coolant distribution through the 758 pressure tubes is controlled by orifices sized according to the radial power distribution to provide a mixed-mean reactor outlet temperature of 721.4 F—equal to that from the hot pressure tube.

To ensure adequate thermal performance of all fuel channels, the "hot channel" method was used to evaluate the thermal performance of the thorium oxide clustered-pin design. The hot channel is composed of the fuel pins and the associated channel in the pressure tube having the most restrictive thermal conditions with respect to geometry, nuclear power peaking, and engineering factors. The hot channel factors used to simulate these conditions are included in Table 2-3. An average power output of 10 kw/ft was used for the hot pin in the hot pressure tube. The GLOBS computer code (described in BAW-393-2¹) was used in calculating the flow and the associated pressure drop required to maintain the hot channel surface temperature at 830 F. The appropriate fin efficiency and the effect of mixing were also incorporated; the efficiency of the fins is 92% because of the high aspect ratio (0.058 inch high/0.030 inch wide).

The MIX-1 computer code was used to determine the effect of mixing between channels. An α (turbulence intensity factor) of 0.02 was used for the design analysis. This value of α is almost a factor of two below corresponding values obtained in tests of unwrapped rods. Because of mixing, the enthalpy rise is about 74% of what it would be for the hot channel without mixing of coolant from adjacent channels. Mixing, as defined by AI-CE,⁹ is given by the following equation:

$$\% \text{ mixing} = 100 \frac{\Delta T_{HC} - \Delta T_{HC}^M}{\Delta T_{HC} - \Delta T_A}$$

where

ΔT_{HC} = temperature rise of the fuel element hot channel without mixing

ΔT_{HC}^M = temperature rise of the fuel element hot channel with mixing

ΔT_A = overall temperature rise of the entire fuel element

Using this definition, a 0.02 α gives 68% mixing for the 37-pin thorium oxide design.

The resultant pressure drop obtained for the hot channel based on the hot channel factors, mixing effects, and the pre-fouling clad surface temperature of 830 F is used to determine the thermal and hydraulic performance of the hot pressure tube. The pin geometry, hot channel, and local radial peaking factors are shown in Figure 2-15. The temperature of the coolant out of the hot pressure tube is 721 F. Figures 2-16 and 2-17 show the calculated coolant and cladding temperatures for the hot channel and the calculated fuel temperatures for the hot pin. The critical heat flux or DNB ratio was calculated for the hot channel and is shown in Figure 2-18. The minimum DNB ratio is 2.6 occurring at 78% of the active channel length downstream of the inlet, based on a conservative saturation temperature of 850 F. A much higher ratio would be obtained if actual pressure distribution along the length of the channel were considered. Critical heat fluxes were calculated using the correlation developed by Core and Sato, which is:¹⁰

$$(Q/A)_C = 100,000 + 408 \Delta T_{sub} (V)^{2/3}$$

where

$(Q/A)_C$ = critical heat flux, Btu/hr-ft²

ΔT_{sub} = saturation temp - bulk temp, F

V = bulk velocity, ft/sec

Fission gas pressure in the hot pin based on release ratio from UO₂ was calculated with TAFYA, a B&W proprietary

computer code. The results of this calculation are shown in Figure 2-19. The predicted fission gas release, which is a function of burnup and temperature as the element is moved through the core, is 23% of that generated. The calculation is based on a total burnup of 29,200 MWD/MT (U + Th) in the hot pin and on the temperature distribution shown in Figure 2-17.

2.4.2. AI-CE Design With Thorium Oxide Fuel Elements

2.4.2.1. Core Description

The preliminary design of a thorium oxide clustered-pin core for the 1000-MWe Uranium HWOCR is described in this section. Design parameters are listed in Table 2-4. As noted in section 2.3.2, the 1000-MWe Uranium HWOCR was defined by extrapolation of the 500-MWe Uranium Demonstration Plant described in AI-CE-MEMO-25.⁴

By definition, thorium cores in the AI-CE plant should produce the same power output as the uranium cores. Since the size and quantity of pressure tubes are also fixed, essentially identical power per pressure tube is required. A 55-pin oxide element is required to achieve this power level because of the limiting heat rating of the oxide fuel. The further requirement that the element fit inside the 4.03-inch-ID pressure tube fixes the pitch and (subject to small variations in fin height) the pin diameter. Thus each fuel element has an OD of 3.970 inches and consists of 55 pins arranged in a triangular pitch array. Each pin has six spiral fins integrally extruded with the clad. The pins are formed from SAP 895 cladding and are filled with Vipac $\text{ThO}_2\text{-UO}_2$ compacted to 88% of theoretical density.

Based on extrapolation of the 500-MWe Uranium Demonstration Plant, the core consists of 548 pressure tubes or fuel channels on a square pitch of 11.0 inches, surrounded by D_2O in a 26-foot, 3-inch-ID moderator tank. Each pressure tube has an ID of 4.030 inches and contains five fuel elements (equivalent active fuel length of 42 inches per element) stacked axially to form a total active fuel length of 17 feet, 6 inches.

As in all the other cores, a one-pass coolant system with bi-directional flow in adjacent pressure tubes is utilized.

Santowax coolant flows through the core at the rate of 121.7×10^6 lb/hr at a total core pressure differential of 227 psi, including headers. The pressure drop across the core is 173 psi, and the total system pressure drop is 327 psi. The coolant transports 3015 MWt of heat while increasing in temperature from 560 to 706.5 F. In addition to the heat transport in the coolant, the reactor generates 158 MWt in the moderator and 10 MWt in the shield for a total output of 3183 MWt. This output produces 1076 MWe at a net plant efficiency of 33.8%.

At the design burnup of 20,000 MWD/T, the idealized-cycle conversion ratio is 0.85 and the U-233 inventory is 1410 kg.

2.4.2.2. Mechanical Design

A cross section of the basic element is shown in Figure 2-20. This element consists of 43 large pins, 12 small pins, and 6 flow blockers. The flow blocker is a circular segment, made of SAP 895, which extends the full length of the element. The small pins are offset toward the large pins as shown in Figure 2-20. Twelve large and ten small outer pins are attached to the ferrule end grid with cap screws to carry the hydraulic, drag, and weight loading. Indexing of one element to another (to prevent rotation) is done in the same manner as for the 37-pin oxide design.

Since the pins are smaller for the 55-pin design than for the 37-pin design, straps will be needed at the 1/3 and 2/3 points to prevent excessive vibration. These straps (0.375 by 0.020) will be made of Inconel X and will surround the element at the outer periphery.

The 43 large pins have a fin OD of 0.501 inch, a clad OD of 0.381 inch, a 0.020-inch wall, and six spiral fins on the outside of the clad. The fins spiral with a helix pitch of 14.3 inches and are 0.060 inch in height by 0.030 inch wide. The 12 smaller pins have a fin OD of 0.429 inch, a clad OD of 0.329 inch, a 0.020-inch wall, and six spiral fins on the outside of the clad with a helix pitch of 14.3 inches. The small pin fins are 0.050 inch in height by 0.030 inch wide.

The overall effective element length is 44 inches with a 42-inch equivalent fuel length. The same type of Zircaloy end grids as used in the 37-pin design would be used here, modified by the pin geometry.

The wall thickness of the SAP pressure tube in this 55-pin design has been increased to that of the AI-CE design in order to accommodate the larger core inlet pressure of the 55-pin design (310 psi vs 229 psi for the 37-pin design).

2.4.2.3. Nuclear Design

Table 2-4 summarizes the results of physics calculations for the thorium oxide clustered-pin fuel element for the AI-CE core. Data given in the table are for the idealized cycle only as defined by an arbitrary fuel loading of 94.5 wt % U-233 and 5.5 wt % U-234. This design has been placed at a slight nuclear disadvantage compared to the thorium-optimized core (section 2.4.1) because of the lower volume fraction of fuel within the pressure tube. The lower fuel volume fraction is necessary to generate the same power per pressure tube as the AI-CE core; however, as indicated in BAW-393-4² and BAW-393-6,³ the conversion ratio decreases with a decrease in the fuel volume fraction. The idealized-cycle conversion ratio for the 55-pin design is 0.85 compared with 0.91 for the 37-pin optimized design.

2.4.2.4. Thermal Design

The same calculation methods as previously described were used for this design. An average power output of 10.24 kw/ft was used for the hot pin in the hot pressure tube. The flow and the associated pressure drop required to maintain the hot channel surface temperature at 830 F were determined with the factors listed in Table 2-4 and with the effect of mixing applied to the hot channel. A 0.02 α (turbulence intensity factor) was used in the MIX-1 computer code. Because of mixing, the enthalpy rise is about 72% of what it would be for the hot channel without mixing of coolant from adjacent channels. On the basis of the AI-CE definition, mixing is 72.6%.

The pin geometry and hot channel and local radial peaking factors are shown in Figure 2-20. The temperature of the coolant out of the hot pressure tube is 706.5 F. Distribution of the flow through the 548 pressure tubes is controlled by orifices sized according to the radial power distribution to provide a mixed mean reactor outlet temperature equal to the outlet temperature (706.5 F) of the hot pressure tube. The hot pressure tube has a nominal power output of 6.43 MWt to

the coolant, and the average pressure tube has a fission power of 5.81 MWt.

The calculated coolant and cladding temperatures for the hot channel are shown in Figure 2-21. The critical heat flux or DNB ratio calculated for the hot channel is shown in Figure 2-22. The minimum DNB ratio is 2.25 occurring at 80% of the active channel length downstream of the inlet based on an 850 F saturation temperature.

2.4.3. Evaluation of Thorium Oxide Designs

The optimization studies indicate that the power costs are slightly higher for the 1000 MWe thorium oxide HWOCR than for the metallic nested-cylinder and the carbide pin designs under the design criteria and assumptions used in developing the designs. However, as previously noted, the design criteria and technical limits should be established carefully and evaluated critically to ensure that different designs can be compared on an equivalent basis. In addition, factors other than economic performance, which is predicted on the basis of a pre-established set of economic ground rules, should be considered in the comparative evaluation to select a reference design.

The principal factors that favor the oxide design are the irradiation stability of the fuel, the adaptability of the Vipac process to remote fabrication, and the relatively advanced stage of development. As a corollary to the requirement for less further development, there is probably less extrapolation and uncertainty involved in the predicted performance of the oxide than in that of the carbide or the metal designs.

Although the power costs for the oxide design are greater than those for the other designs, the difference is relatively small and probably within the uncertainty band. However, there are inherent characteristics which are reflected as economic penalties. These characteristics are the limited pin heat rating and the cumulative effects of several factors that are minor when considered alone. Relative to the thorium metal nested-cylinder design, the most significant factors are the less efficient thermal performance of pin-type cores (as discussed in section 2.3.3) and the limited power per pressure tube with thoria-urania pins. Likewise, the most important consideration in comparing the oxide with thorium monocarbide is the lower thermal conductivity of thoria-

urania, which is reflected in higher fuel temperatures and fission gas release rates.

The performance of the thorium oxide design is limited by fission gas pressure buildup. The use of SAP cladding and the low external pressure on the fuel pins make fission gas pressure much more critical in the HWOCR than in previous applications of thorium oxide (e.g., Indian Point, which has stainless steel cladding and a 1500-psi operating pressure). Since the fission gas pressure buildup is dependent upon the pin heat rating and the burnup, 10 kw/ft (average in hot pin) and 20,000 MWD/T (average) were selected as a combination of values for near-optimum economic performance within fission gas pressure limitations.

The fuel temperature is a function of the linear heat rate (kw/ft) and the thermal conductivity. Therefore, thermal conductivity exerts its influence primarily by limiting the specific power. Economic performance depends not only on high specific power but also on the method of achieving the increase—by increasing pin heat rating or by reducing pin diameter. Both fuel cycle and plant capital costs decrease rapidly with increasing specific power when the increase is achieved by raising the pin heat rating. However, the low oxide conductivity in effect establishes the total length of fuel pins, which must be approximately 50% greater for oxide than for carbide and, to a large extent determines the reactor size. The reactor size and the specific power are also dependent, to some extent, on the fuel pin diameter and the assembly dimensions. Because fabrication costs increase and the conversion ratio decreases with a decrease in pin diameter, the fuel cycle cost is a minimum with pin diameters near 0.50 inch. In addition, pressure tube radial peaking factors increase with increasing fuel element diameter and tend to reduce the diameter of the optimum pressure tube assembly. Thus the low heat rating, large optimum pin diameter, and small optimum pressure tube assembly lead to a thorium oxide core design with a large number of pressure tube assemblies. Since the reactor is larger, the capital costs for the thorium-urania pin cores are higher than those for the thorium carbide (ThC) clustered-pin or metallic nested-cylinder cores.

Because of the importance of thoria-urania thermal conductivity, considerable effort was made to review and evaluate existing data. The results of this evaluation were applied in the design effort. However, R&D results are definitely required to confirm the values used or to define more clearly the limits on operation. While the possibility of favorable results from an R&D program exists, the probability of significant increases in pin heat rating (due to more precise thermal conductivity data) appears minimal. Assessment of power cost variations with pin heat rating indicates a change of about 0.01 mill/kwhr per percent for small changes in heat rating from the design value of 10 kw/ft (average in the hot pin). The rate of change decreases at higher heat ratings as indicated by the carbide pin performance.

In addition to its importance in establishing specific power in optimum pin sizes, the thermal conductivity affects the fuel cycle cost indirectly because of its part in limiting the fuel burnup. For example, fuel cycle costs for the thoria-urania pin cluster design can be improved about 0.10 mill/kwhr by increasing the fuel burnup to the economic optimum. As stated, the burnup limit of 20,000 MWD/T (average) for these designs is controlled by fission gas pressure buildup; consequently, it is dependent on fuel temperature, which is directly related to thermal conductivity of the fuel.

While the thermal conductivity data for Vipac thoria-urania fuel are inadequate for firm design, the fission gas release rates are even more uncertain. For example, UO_2 release data and analytical models have been used in predicting fission gas pressure for design purposes. The few data points available from ORNL irradiations of thorium oxide indicate that physical changes which develop during irradiations and corresponding gas release rates will occur at heat ratings 10 to 40% higher than with UO_2 . Vibratory compacted ThO_2 - UO_2 rods have been operated to 20,000 MWD/T at heat ratings up to 26 kw/ft with no evidence of swelling and with less than 30% fission gas release. These trends demonstrate the irradiation stability of thoria-urania and indicate that the possibility of extending burnup beyond the specified 20,000 MWD/T is considerably greater for the oxide than for the other fuel materials. Demonstration of fission gas release characteristics more favorable than those assumed in the design would probably permit a 50% increase in

burnup before other irradiation characteristics began to limit performance. Such an increase could improve the economic performance by 0.05 to 0.10 mill/kwhr.

Another potential problem is the possibility that the SAP cladding cannot withstand the vibration required to attain the fuel density proposed by the Vipac process. However, should this potential problem materialize, it would be detected early in the development program and with minimum expense. Furthermore, alternate routes to use thorium-uranium fuels are provided by the use of pellets. The remote fabrication costs would probably be somewhat higher, but such advantages as higher effective conductivity and lower fission gas release should partially offset any increase.

Naturally, minimized fouling and compatibility of the low-ductility SAP clad with the organic environment must be demonstrated under HWOGR operating conditions. The AI-CE program should answer these questions satisfactorily. The performance of the SAP-clad Vipac thorium-uranium fuel in an HWOGR environment should be verified by the program outlined in section 3.

As noted in section 2.4.2, the oxide-pin element is basically compatible with the AI-CE HWOGR design, but pin diameters smaller than the economic optimum are required in a 55-pin element to equal the power output of the 37-pin uranium carbide fuel element. The requirement for a larger number of smaller pins dictates a somewhat different design from the AI-CE design, and some independent engineering tests will be required. Fabrication costs are also higher for the smaller-diameter pins.

In summary, the economic performance of the thorium oxide core is slightly less favorable than the nested-cylinder core and about equal to that of the carbide element at the design conditions. However, the difference in power cost is small, and the uncertainty of performance predictions for the oxide core may be less than for the other designs. At the same time, the parameters most likely to be affected by R&D results—specific power or pin heat rating and fission-gas-limited burnup—are as important economically as the corresponding areas of concern in the nested-cylinder design—channel tolerances and fuel-swelling-limited burnup.

Development costs for the thorium oxide core will be less than those for the other types of fuel materials, and the development can be completed on a shorter schedule. Most of the engineering test results from the UC clustered-pin fuel element should be applicable to the oxide element.

The conclusion from the draft HWO CR Evaluation Report⁸ provides an independent summary of the evaluation of a similar thorium oxide pin-type core:⁸

"The SAP-clad vibratory compacted element shows promise for HWO CR application; however, additional data and testing are required to show that the fuel element will operate satisfactorily under the proposed conditions. Particularly, experiments must be performed demonstrating that the effect of burnup in SAP-clad, vibratorily compacted fuel is not detrimental to the cladding. Also, it must be shown experimentally that SAP cladding is not adversely affected by the vibratory compaction operation."

2.5. Thorium Carbide Core Design

2.5.1. Core Description

The thorium carbide clustered-pin core design for the 1000-MWe HWO CR is described in this section. Table 2-5 lists design parameters. The thorium-optimized core is identical in size and configuration to the 1000-MWe HWO CR that B&W extrapolated from the 500-MWe HWO CR Demonstration Plant described in AI-CE-MEMO-25.⁴ Therefore, the thorium-optimized core and the thorium core for the AI-CE HWO CR are the same. The fuel element has 37 pins in a 4.03-inch-ID pressure tube; the pins are composed of SAP 895 cladding and filled with thorium-uranium monocarbide slugs.

The core has 548 pressure tubes on a square pitch of 11.0 inches, surrounded by D₂O in a 26.25-foot-ID moderator tank. Each pressure tube has an ID of 4.030 inches and contains five fuel elements (equivalent active fuel length of 42.66 inches per element) stacked axially to form a total active fuel length of 17 feet, 9.4 inches.

Santowax coolant flows through the core at the rate of 113×10^6 lb/hr at a total core pressure differential of 209 psi including header and shield plug losses. The pressure drop across the core is 155 psi,

and the total system pressure drop is 309 psi. The coolant transports 2984 MW of heat while increasing its temperature from 560 to 716.2 F. In addition to the heat transferred to the coolant, the reactor generates 157 MWt in the moderator and 10 MWt in the shield for a total output of 3151 MWt. This output will produce 1076 MWe at a net plant efficiency of 34.1%. The power removal capability of an individual pressure tube is 6.69 MWt with a flux tilt factor of 1.05.

2.5.2. Mechanical Design

The pin-type carbide fuel element mechanical design is the same as the oxide design (described in section 2.4.1.2 and shown in Figure 2-13) except for the following differences:

1. The large pin cladding OD is 0.495 inch (oxide is 0.515), the ID is 0.451 inch (oxide is 0.471), and the fins are 0.068 inch high (oxide is 0.058). Because linear heat ratings of the carbide pins are about 50% higher than those of the oxide pins, more flow area is required for adequate cooling; therefore the basic fin height for the carbide design is greater.
2. The fuel is ThC-UC in the form of slugs rather than Vipac $\text{ThO}_2\text{-UO}_2$.
3. The void length is 0.600 and 0.200 inch on the large and small pins (oxide is 1.3 and 0.43).
4. The overall element length is 44.12 inches (oxide is 44.82). The reduced void space permits a shorter element.
5. Equivalent nonfuel length is 1.455 inches (oxide is 2.155). The reduced void space permits a shorter nonfuel length.

The fuel for this element is thorium-uranium monocarbide cast into slugs approximately 3 inches long. One end of the pellet is dished to a depth of 0.040 inch with a land width of 0.020 inch. The resultant density including gap, dish, and fabrication allowances is 10.1 gm/cm³. The slugs are finish-ground to give a diametral clearance of 0.005 inch on both the large and small fuel pins. A tolerance of ± 0.0005 inch on both the OD of the slugs and the ID of the fuel tube gives a diametral clearance of 0.004 to 0.006 inch, which is sufficient to prevent

adverse extension of the clad at operating conditions and which permits the pellets to be loaded in a standard manner. The ThC includes 5 wt % (nominal) carbon.

No work has been done with the combination of SAP cladding and ThC-UC fuel, but based on UC performance, no incompatibility is expected between the cladding and the fuel as a result of thermal and irradiation growth. Since ThC is highly reactive with moisture, remote recycle fabrication will require a dry, inert atmosphere or the addition of a stabilizing agent to the fuel material.

2.5.3. Nuclear Design

Table 2-5 summarizes the results of physics calculations for the 37-pin thorium carbide fuel element in a 4.030-inch-ID pressure tube. The physics methods used are described in BAW-393-4.² Results given in Table 2-5 include data for the U-233 idealized cycle and for U-235 startup and recycle. THERMOS-corrected, local-radial power peaking ratios are given in Table 2-5 for both the idealized cycle and the initial U-235 cycle at time zero. Plutonium startup calculations were not made for the carbide core. The following power peaking factors are assumed on the basis of calculations for alternate core designs:

Local axial $P/\bar{P}$	1.17
Gross radial $P/\bar{P}$	1.17
Gross axial $P/\bar{P}$	1.30

Local radial peaking factors were calculated and the results are reported in Table 2-5.

The validity of the standard physics lifetime model for determining the equilibrium cycle of a continuously refueled and recycled core is discussed in section 2.3.1.3.

2.5.4. Thermal Design

The thermal and hydraulic parameters for the 37-pin thorium carbide design are listed in Table 2-5. The calculation methods and hot channel factors described in section 2.4.1.4 were also used for this design. An average power output of 14.3 kw/ft was used for the hot pin. The flow and the associated pressure drop required to maintain the hot channel prefouling surface temperature at 830 F were based on the

geometry, the local radial peaking factors, and the hot channel location shown in Figure 2-23.

The effect of mixing was applied to the hot channel, and a turbulence intensity factor (α) of 0.02 was used in the MIX-1 computer code. Due to mixing, the enthalpy rise is about 77% of what it would be for the hot channel without mixing of coolant from adjacent channels. On the basis of the AI-CE definition, mixing is 66.1%.

The high aspect ratio (0.068 inch high/0.030 inch wide) of the fins provides a fin efficiency of 87%.

Due to insufficient data on thorium monocarbide properties and complete absence of information on irradiation performance, the fuel temperatures, fuel swelling, and fission gas release were based on uranium monocarbide data given in AI-CE-MEMO-25.⁴

The temperature of the coolant out of the hot pressure tube is 716.2 F. The distribution of flow through the 548 pressure tubes is controlled by orifices sized according to the radial power distribution to provide a mixed mean reactor outlet temperature equal to the outlet temperature of the hot pressure tube (716.2 F). The hot pressure tube has a nominal power output of 6.37 MWt to the coolant, and the average pressure tube has a fission power of 5.75 MWt.

The calculated coolant and cladding temperatures for the hot channel are shown in Figure 2-24. The critical heat flux or DNB ratio calculated for the hot channel is shown in Figure 2-25. The minimum DNB ratio is 2.34 occurring at 80% of the active channel length downstream of the inlet based on an 850 F saturation temperature.

4.5.5. Evaluation of Thorium Carbide Core Design

The relative economic performance of the carbide pin design falls below that of the metallic nested-cylinder and slightly above that of the thoria-urania-pin designs. This position stems from the combination and basic characteristics of the fuel material, the coolant, and the reactor concept.

The important design characteristics of carbide fuel are its relatively high mass of thorium per unit volume and its high thermal conductivity. Compared with oxide fuel, the higher mass of fertile material contained in any given fuel element configuration leads to slightly

higher conversion ratios. The higher thermal conductivity allows the fuel pin heat rating and the average specific power to be increased about 50% above the maximum design values using oxide fuel. The slightly higher conversion ratio reduces the burnup cost, and the higher specific power reduces both capital costs and inventory charges. Thus at first glance, it appears that power costs should be lower with carbide than with oxide cores. However, there are opposing trends that offset the potential advantages to a large extent.

In order to realize the advantage provided by high thermal conductivity, the carbide should be in solid form, rather than vibratory-packed powder. This requirement further complicates the remote fabrication and reprocessing that is essential for the long-term recycle of bred fuel. Consequently, fabrication and reprocessing costs are higher for carbide than for Vipac oxide fuel, offsetting at least to some extent the gain in inventory and burnup costs. This higher fabrication cost increases the economic optimum burnup, which may not be achievable because of fuel swelling and its effect on the low-ductility SAP cladding.

The higher allowable pin heat rating and specific power that provide the chief economic advantages for carbide also necessitate more stringent thermal-hydraulic performance than would be necessary for the lower-heat-flux oxide-pin design. Either larger coolant flow areas or much higher core pressure differential and coolant velocities are required to remove the higher-power-per-unit core volume. The larger coolant flow area naturally reduces the fertile material volume fraction and partially offsets the potentially higher conversion ratio capability. The use of additional fins to lower the heat flux has the same effect.

The larger pressure differentials, together with lower outlet temperatures that result from higher heat fluxes, decrease the net plant efficiency and tend to reduce the initial benefits from a smaller-size, lower-capital-cost core. Since a rather large reduction in core capital cost is required to offset a relatively small change in plant efficiency, the economic benefits of the smaller core, while real, are smaller than would at first be expected. Likewise, the fuel cycle costs are subject to similar offsetting effects. While the higher specific power

reduces the carbide inventory cost, fabrication costs increase and essentially offset the saving in inventory. The higher specific power, together with the slightly larger coolant flow area required in the pressure tube, results in a conversion ratio about equal to that obtainable with the lower-power-density oxide designs. Therefore, there is practically no saving in the burnup cost. Thus, with fuel cycle costs comparable to those of the oxide designs and capital costs comparable to those of the nested-cylinder designs, the economic performance of the carbide designs is intermediate between the other two.

As noted in section 4, the fabrication cost is less certain for the carbide fuel elements than for the other two fuel materials. This uncertainty is a result of the less-developed status of ThC in that the fabrication and reprocessing requirements have not been completely established. Since the fabrication cost is based on remote operation of the UC fabrication plant, and reprocessing cost is based on oxide reprocessing cost, the values used are probably low, and at best, the fabrication cost estimate has a relatively large uncertainty.

The basic lack of sufficient data to adequately judge the potential of thorium carbide is a major deterrent to selection of this fuel material for the reference design. As noted in section 2.5.4, the fuel temperature, fuel swelling, and fission gas release were assumed to be similar to those of UC fuel. While meager data would support different values (as noted in section 3.4), there is not enough confirmed information to justify the additional design effort required. Furthermore, if the use of UC properties did not lead to favorable performance, it is doubtful that more effort to define the real properties would be productive.

The reactive nature of ThC (also discussed in section 3.4) would lead to fabrication difficulties unless a stabilizing additive could be developed and proved. Lack of irradiation performance data is also of paramount concern. Thus, extensive fuel characterization would be required before acceptability of the ThC could be assured.

The ThC element is perfectly adaptable to the AI-CE plant, at least on the basis of the design assumptions; however, the changes that might be required as more precise information becomes available cannot be predicted quantitatively. In summary, lack of economic incentive, uncertain fuel fabrication-reprocessing cost, the reactive nature of ThC, and the meager irradiation and properties data prohibit a factual evaluation of this design.

Table 2-1. Core Design Data for Four-Cylinder Thorium
Metal Fuel Element (With Center Moderator)
in a 5.0-Inch-ID Pressure Tube

I. Reactor Data

General

Moderator	D ₂ O
Coolant	Santowax OM-10% HB
Fuel	Thorium-U-233 (low enriched)
Fission power, MWt	3187
Thermal power to coolant, MWt	3019
Thermal power to moderator, MWt	158
Net electric power, MWe	1076
Number of coolant passes	1
Active fuel length, ft-in. (5 × active fuel length/element)	18-9
Active core length, ft-in. (fuel to fuel for 5 elements)	19-2.2
Overall stack length, ft-in. (end plate to end plate)	19-3.5
Maximum peak-to-average power ratios	
Gross radial	1.17
Local radial	1.64
Gross axial	1.30
Local axial	1.10
Flux tilt	1.05
Pitch of fuel channel assemblies, in.	11.6
D ₂ O inventory in calandria, lb	440,705
D ₂ O cover gas	Helium

Fuel Channel Assembly

	Inner tube (in-core section)	Outer tube (in-core section)	Lower fuel channel support tube	Upper fuel channel support tube
Material	SAP	Zr-2	SS	SS
OD, in.	5.262	6.130	7.340	7.340
Wall thickness, in.	0.131	0.080	0.25	0.25

Table 2-1. (Cont'd)

	<u>Inner tube (in-core section)</u>	<u>Outer tube (in-core section)</u>	<u>Lower fuel channel support tube</u>	<u>Upper fuel channel support tube</u>
Length, ft-in.	21-4	21-4	3-7	3-7
Overall fuel channel assembly length, ft-in.				37-4
Insulation between inner and outer tubes				CO ₂
Height of fuel channel assembly extending above top shield, ft-in.				5-3
Height of fuel channel assembly extending below bottom shield, ft-in.				4-9
Total number of fuel channels in core				324
Direction of coolant flow in fuel channels				Up & down
Direction of fuel flow in fuel channels				Up & down
Method of refueling			On-line, by machine, from top & bottom	
How fuel channel is isolated			Manually operated valves	
<u>Moderator Tank</u>				
Material				SS
ID, ft-in.				21-7
Wall thickness, in.				1
Inside shield length, ft-in.				21-4
Top shield height, ft-in.				3-0
Top shield diameter, ft-in.				23-9
Bottom shield height, ft-in.				3-0
Bottom shield diameter, ft-in.				23-9
<u>Piping</u>				
Total number of pipes from fuel channel assemblies to header (each end)				324
Joint between fuel channel assemblies and pipes to headers				Mechanical

Table 2-1. (Cont'd)

Piping material	Carbon steel
Pipes to headers, size, in.	3-1/2 Sch 40

II. Fuel Element Data

General

Drawing number A30461 E	Figure 2-1
Fuel assemblies/fuel channel	5
Fuel assemblies/reactor	1620
Fuel cylinders/assembly	4
Refueling rate, assemblies/day @ 20,000 MWD/T	2.0
Fuel content/assembly, kg (U + Th)	61.6
Fuel element weight, lb	168
Materials composition inside homogenized fuel channel, vol %	
Fuel	35.6
Structural material (all Zr)	16.0
Coolant	48.4

Dimensions (Hot)

Active fuel length/assembly, in.	
Cyl 1	45
Cyl 2	45
Cyl 3	45
Cyl 4	44.55
Overall length (effective), in.	46.3
Zr-2 center moderator can diameter, in.	1.866
Zr-4 cladding thickness, in. (typical for all cylinders)	0.025
Fuel thickness, in.	
Cyl 1	0.242
Cyl 2	0.205
Cyl 3	0.150
Cyl 4	0.100
Coolant gap thickness, in.	
Cyl 1 - center can	0.110
Cyl 1 - 2	0.150
Cyl 2 - 3	0.150

Table 2-1. (Cont'd)

Cyl 3 - 4	0.150
Cyl 4 - pressure tube	0.110

III. Thermal and Hydraulic Data

General

Core inlet temperature, F	560
Core mixed mean outlet temperature, F	708.4
Total coolant flow, lb/hr	119.0×10^6
Core inlet pressure, psi	322
Total core pressure drop (including header), psi	239
Minimum ratio, critical flux to actual flux	2.4

Core

General

Number of pressure tubes	324
Power to coolant, MWt	3019
Pressure drop, active core including head, psi	185
Pressure drop, headers, psi	54
Fission power/avg pressure tube, MWt	9.8
Cylinders/element	4

Fuel Tubes

Nominal Pressure Tube^(a)

Power/pressure tube, MWt	10.9
Mixed mean outlet temp, F	708

	<u>Cylinder</u>			
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>
Max surface temp, F				
ID	772	776	781	779
OD	773	777	782	771

^(a) Nominal pressure tube has nominal dimensions and nominal power peaking factors.

γ

Table 2-1. (Cont'd)

	Cylinder			
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>
Max heat flux, Btu/hr-ft ² × 10 ⁻⁵				
ID	3.44	3.49	3.57	3.63
OD	3.41	3.44	3.58	3.54
Max fuel temp, F	883	880	875	856
Peak-to-average power ratio	0.649	0.774	1.092	1.641

Hot Pressure Tube^(b)

Power/pressure tube, MWt	11.65
Mixed mean outlet temperature, F	731

	Cylinder			
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>
Max surface temp, F				
ID	782	787	796	820
OD	783	788	813	815
Max heat flux, Btu/hr-ft ² × 10 ⁻⁵				
ID	3.62	3.66	3.78	4.00
OD	3.58	3.60	3.71	3.88
Max fuel temp, F	902	899	909	911
Peak-to-average power ratio	0.672	0.801	1.033	1.567

(b) Hot pressure tube has maximum power peaking factors and worst tolerances for geometry and power generation. Dimensional variations caused by fabrication tolerances and fuel growth are incorporated in the fuel element geometry used in the thermal analysis. The outer cylinder is assumed to have a power generation factor of 1.05.

Table 2-1. (Cont'd)

Coolant Channels

	Channel				
	1	2	3	4	5
Nominal Pressure Tube					
Channel thickness, in.	0.110	0.150	0.150	0.150	0.110
Channel flow, lb/hr $\times 10^{-3}$	35.3	82.4	106.0	126.5	87.3
Avg velocity, fps	38.1	45.9	45.9	45.9	38.0
Channel exit temp, F	696	711	714	716	687
Hot Pressure Tube					
Channel thickness, in.	0.110	0.150	0.150	0.1365	0.0965
Channel flow, lb/hr $\times 10^{-3}$	35.2	82.4	106.0	109.1	70.9
Avg velocity, fps	38.1	46.0	46.0	44.0	35.5
Channel exit temp, F	703	719	724	751	730

IV. Nuclear DataU-235 Batch Recycle for Thorium Metal Nested-Cylinder Core

	Cycle 1		Cycle 2		Cycle 3		Cycle 4		Cycle 5		Cycle 6	
	In	Out	In	Out	In	Out	In	Out	In	Out	In	Out
Th-232	300.337	294.595	299.987	294.239	299.450	293.760	298.913	293.281	248.439	292.739	298.034	292.308
U-233	--	2.893	3.206	3.860	4.167	4.150	4.450	4.241	4.537	4.270	4.563	4.279
U-234	--	0.296	0.292	0.678	0.668	1.010	0.995	1.277	1.258	1.489	1.467	1.654
U-235	7.399	2.281	3.397	1.109	2.242	0.825	1.963	0.784	1.922	0.801	1.939	0.832
U-236	--	0.831	0.819	1.157	1.140	1.334	1.314	1.471	1.449	1.599	1.575	1.724
U-238	0.562	0.520	0.599	0.554	0.633	0.587	0.665	0.617	0.695	0.645	0.722	0.670
Pa-233	--	0.366	--	0.374	--	0.372	--	0.369	--	0.366	--	0.363
Pu-239, Pu-241	--	0.013	--	0.014	--	0.015	--	0.015	--	0.016	--	0.017
Total	308.3		308.3		308.3		308.3		308.3		308.3	
Full power days		656		636		625		627		635		638
MWD/T metal		20,214		19,598		19,259		19,320		19,567		19,659
C.R.		0.7597		0.8263		0.8498		0.8558		0.8571		0.8572
Makeup												
Th-232		9.811		9.625		9.559		9.557		9.686		
U-235		1.150		1.150		1.150		1.150		1.150		
U-238		0.087		0.087		0.087		0.087		0.087		

Notes: 1. Run at 9.5 MWt/pressure tube.
 2. Masses are kg/pressure tube.

3. 180-day cooling for Pa decay.
 4. Total losses of 1.5% during recycle.

Table 2-1. (Cont'd)

Continuous Refueling of Thorium Metal Nested-Cylinder Core

Batch No. Code

x	y
---	---

x = original batch No.

y = times recycled

(i.e., #11 is batch 10 after first recycle)

	Batch No.																					
	10(c)	20(c)	30(c)	40(c)	50(c)	60	70	80	11	21	31	41	51	61	71	81	12	22	32	42	52	62
FPD in	0	0	0	0	0	249	379	509	639	769	899	1029	1159	1289	1419	1549	1679	1809	1939	2069	2199	2329
out	249	379	509	639	769	899	1029	1159	1289	1419	1549	1679	1809	1939	2069	2199	2329	2459	2589	2719	2849	2979
MWD	2365	3601	4836	6071	7306	8541	9776	11011	12246	13481	14716	15951	17186	18421	19656	20891	22126	23361	24596	25831	27066	28301
MWD/T	7674	11,680	15,686	19,692	23,698	27,704	31,710	35,716	39,722	43,728	47,734	51,740	55,746	59,752	63,758	67,764	71,770	75,776	79,782	83,788	87,794	91,800
C. 3.	0.7181	0.7408	0.7618	0.7813	0.7991	0.7767	0.7252	0.7565	0.8114	0.8461	0.8342	0.8138	0.8315	0.8423	0.8460	0.8360	0.8416	0.8478	0.8603	0.8610	0.8599	0.8566
FPD put back in	639	769	899	1029	1159	1289	1419	1549	1679	1809	1939	2069	2199	2329	2459	2589	2719	2849	2979	3109	3239	3369
Masses in																						
E. wt % U-233	--	--	--	--	--	--	--	--	0.5469	0.7545	0.9173	1.0426	1.1385	1.0484	1.0115	1.0338	1.2130	1.2812	1.3215	1.351	1.382	1.359
E. wt % U-235	2.10	2.30	2.30	2.30	2.30	2.325	2.550	2.410	1.641	1.307	1.1940	1.1522	0.9955	1.0393	1.0604	1.0767	0.8874	0.8024	0.7230	0.6947	0.6740	0.7051
kg Th	300.627	300.627	300.627	300.627	300.627	300.567	300.305	300.567	300.673	300.337	299.831	299.831	300.167	300.167	300.061	299.831	299.767	299.721	299.601	299.431	299.555	299.431
U-233	--	--	--	--	--	--	--	--	1.686	2.326	2.828	3.214	3.510	3.232	3.118	3.187	3.740	3.950	4.074	4.168	4.261	4.189
U-234	--	--	--	--	--	--	--	--	0.066	0.131	0.209	0.294	0.383	0.297	0.270	0.287	0.437	0.530	0.602	0.670	0.757	0.687
U-235	7.087	7.087	7.087	7.087	7.087	7.146	7.866	7.430	5.058	4.028	3.681	3.552	3.069	3.204	3.269	3.319	2.736	2.474	2.229	2.142	2.078	2.174
U-236	--	--	--	--	--	--	--	--	0.424	0.578	0.697	0.786	0.851	0.798	0.847	0.817	0.961	0.995	1.063	1.128	1.104	1.134
U-238	0.538	0.538	0.538	0.538	0.538	0.545	0.598	0.565	0.560	0.548	0.571	0.600	0.590	0.577	0.601	0.594	0.603	0.596	0.603	0.623	0.620	0.614
Total kg	308.25	308.25	308.25	308.25	308.25	308.32	308.53	308.30	308.36	308.28	308.32	308.28	308.23	308.28	308.27	308.27	308.31	308.29	308.33	308.29	308.32	308.23
Makeup																						
kg U-235	--	--	--	--	--	--	--	--	0.598	0.532	0.950	1.416	1.391	1.069	0.716	1.030	1.237	1.304	1.110	1.005	1.086	1.177
wt %	--	--	--	--	--	--	--	--	0.1940	0.1726	0.3081	0.4593	0.4513	0.3467	0.2323	0.3341	0.4012	0.4230	0.3600	0.3259	0.3523	0.3818
Masses out																						
kg Th	298.528	279.349	296.329	294.874	293.591	294.698	294.624	294.655	294.649	294.569	294.371	294.052	293.969	294.193	294.154	294.125	293.927	293.853	293.748	293.646	293.507	293.633
U-233	1.379	2.014	2.512	2.895	3.186	2.912	2.815	2.875	3.424	3.630	3.762	3.865	3.957	3.879	3.847	3.864	4.026	4.089	4.125	4.152	4.181	4.159
U-234	0.067	0.134	0.212	0.298	0.389	0.302	0.274	0.291	0.444	0.538	0.611	0.680	0.769	0.697	0.674	0.685	0.824	0.907	0.972	1.028	1.097	1.040
U-235	4.328	3.549	2.773	2.169	1.704	2.168	2.592	2.324	1.522	1.188	1.136	1.154	1.007	1.012	1.018	1.051	0.911	0.844	0.777	0.764	0.760	0.779
U-236	0.430	0.587	0.708	0.798	0.864	0.810	0.860	0.829	0.976	1.010	1.079	1.145	1.155	1.121	1.175	1.151	1.221	1.227	1.263	1.315	1.318	1.296
U-238	0.523	0.515	0.507	0.499	0.491	0.504	0.555	0.524	0.517	0.505	0.527	0.555	0.545	0.533	0.555	0.549	0.557	0.550	0.557	0.573	0.567	0.599
Pa-233	0.337	0.351	0.363	0.370	0.381	0.372	0.354	0.365	0.377	0.384	0.378	0.370	0.373	0.378	0.380	0.377	0.374	0.376	0.375	0.374	0.374	0.374
Fiss. Pu	0.009	0.011	0.012	0.012	0.012	0.012	0.013	0.013	0.013	0.011	0.013	0.014	0.014	0.013	0.014	0.014	0.014	0.014	0.014	0.014	0.014	0.014

(c) Batches 10, 20, 30, 40, and 50 were loaded together at time zero. At 249 days, #10 was removed and #60 was put in; at 379 days, #20 was removed, and #70 was put in; etc.

- Notes: 1. Run at 9.5 MW/pressure tube. 3. 365-day minimum holdup for recycle.
 2. Core runs 7674 MWD/T as a batch core initially, then enters continuous fueling. 4. Each "batch" represents 1/5 of the core, although masses are in kg/pressure tube.

Table 2-1. (Cont'd)

Plutonium Batch Recycle of 4-Ring No. 5 Nested Cylinder Element

	Mass balance, kg/pressure tube											
	Cycle 1		Cycle 2		Cycle 3		Cycle 4		Cycle 5		Cycle 6	
	In	Out	In	Out	In	Out	In	Out	In	Out	In	Out
Th-232	298.525	294.231	297.500	293.013	297.500	293.017	297.393	292.634	297.393	292.742	297.163	292.434
U-233	--	2.461	2.717	3.739	3.977	4.245	4.484	4.444	4.684	4.503	4.746	4.522
U-234	--	0.200	0.197	0.484	0.477	0.764	0.753	1.027	1.012	1.242	1.223	1.421
U-235	--	0.022	0.022	0.081	0.080	0.145	0.143	0.207	0.204	0.258	0.254	0.300
U-236	--	0.002	0.002	0.010	0.010	0.028	0.028	0.056	0.055	0.092	0.091	0.131
Pu-239	6.225	1.273	3.028	0.469	1.707	0.229	1.380	0.150	1.238	0.135	1.287	0.134
Pu-240	3.396	3.079	4.000	2.828	3.465	2.237	2.833	1.735	2.303	1.426	2.035	1.272
Pu-241	1.497	1.060	1.471	0.919	1.204	0.725	0.992	0.569	0.822	0.477	0.748	0.430
Pu-242	0.672	1.163	1.337	1.799	1.906	2.228	2.320	2.525	2.605	2.710	2.794	2.847
Pa-233	--	0.300	--	0.302	--	0.307	--	0.314	--	0.318	--	0.320
Total	310.3		310.3		310.3		310.3		310.3		310.3	
Full power days		653		656		627		650		624		634
MWD/T metal		19,991		20,084		19,195		19,899		19,103		19,407
C.R.		0.7182		0.7974		0.8316		0.8342		0.8326		0.8251
Makeup												
Th-232		7.682		8.882		8.771		9.149		8.812		
Pu-239		1.774		1.245		1.154		1.090		1.154		
Pu-240		0.967		0.679		0.630		0.594		0.630		
Pu-241		0.427		0.299		0.278		0.262		0.278		
Pu-242		0.191		0.134		0.125		0.118		0.125		

Notes: 1. 9.5 MWt/tube.

2. 180-day Pu decay, total losses are 1.5%.

3. Pu feed is HWOCR discharge: Pu-239 = 0.528, 240 = 0.288, 241 = 0.127, 242 = 0.057.

Thorium Metal Nested-Cylinder Core — Idealized Cycle

U-233, wt %	1.75	1.80	1.875
k_{eff}	1.1245	1.1385	1.1584
Days	374	483	625
Tons	0.30755	0.30754	0.30773
MWD/T	11,553	14,920	19,295
C.R.	0.9232	0.9147	0.9034
MW/pressure tube	9.5	9.5	9.5
Fuel fraction	0.3500	0.3500	0.3500
Fuel/cell	0.0511	0.0511	0.0511
Watts/gram	30.89	30.89	30.89

Masses

Time 0

Th	301.869	301.699	301.639
U-233	5.382	5.536	5.769
U-234	0.297	0.306	0.319

Final

Th	298.185	296.991	295.647
U-233	4.607	4.608	4.607
U-234	0.639	0.732	0.844
U-235	0.074	0.097	0.127
U-236	0.005	0.009	0.015
Pa	0.393	0.391	0.387

Table 2-1. (Cont'd)

Thorium Metal (Optimized) Core Neutron Balances

	U-233 idealized cycle		U-235 batch recycle				Plutonium batch recycle			
			Cycle 1		Cycle 6		Cycle 1		Cycle 6	
	BOL	EOL	BOL	EOL	BOL	EOL	BOL	EOL	BOL	EOL
<u>Absorptions</u>										
H	0.0286	0.0289	0.0256	0.0279	0.0261	0.0273	0.0148	0.0217	0.0201	0.0253
D	.0023	.0023	.0022	.0022	.0022	.0022	.0017	.0021	.0019	.0021
C	.0004	.0004	.0003	.0003	.0003	.0003	.0002	.0003	.0002	.0003
O	.0042	.0042	.0042	.0042	.0042	.0042	.0041	.0041	.0041	.0042
Al	.0126	.0127	.0117	.0125	.0118	.0122	.0085	.0114	.0100	.0118
Fission products	--	.0483	--	.0505	--	.0471	--	.0509	--	.0471
Zr	.0181	.0183	.0170	.0180	.0172	.0177	.0128	.0159	.0148	.0169
Th-232	.3953	.3908	.3533	.3745	.3605	.3673	.2042	.2758	.2652	.3161
U-233	.5078	.4033	--	.2611	.3656	.3558	--	.1620	.2584	.3079
U-234	.0075	.0208	--	.0078	.0260	.0306	--	.0047	.0170	.0240
U-235	--	.0128	.5601	.1716	.1529	.0636	--	.0018	.0137	.0212
U-236	--	.0001	--	.0054	.0072	.0078	--	--	.0006	.0009
U-238	--	--	.0028	.0024	.0034	.0031	--	--	--	--
Pu-239	--	--	--	.0025	--	.0031	.4921	.1265	.1675	.0181
Pu-240	--	--	--	.0008	--	.0010	.1005	.1136	.0809	.0555
Pu-241	--	--	--	.0006	--	.0007	.1325	.1412	.1040	.0720
Pu-242	--	--	--	--	--	--	.0081	.0152	.0202	.0228
Xe	--	.0217	--	.0213	--	.0213	--	.0164	--	.0196
Sm	--	.0036	--	.0047	--	.0040	--	.0081	--	.0052
Pa-233	--	.0086	--	.0085	--	.0078	--	.0061	--	.0065
Np-239	--	--	--	--	--	--	--	--	--	--
Leakage	0.0233	0.0233	0.0228	0.0231	0.0227	0.0229	0.0205	0.0222	0.0214	0.0225
Total	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00
<u>Productions</u>										
Th-232	0.0072	0.0071	0.0072	0.0071	0.0071	0.0070	0.0070	0.0071	0.0071	0.0070
U-233	1.1510	.9145	--	.5903	.8278	.8060	--	.3649	.5826	.6965
U-234	0.0002	.0005	--	.0002	.0008	.0009	--	--	.0007	.0008
U-235	--	.0258	1.1290	.3463	.3075	.1278	--	.0036	.0272	.0424
U-236	--	--	--	.0002	.0004	.0004	--	--	--	--
U-238	--	--	0.0001	--	0.0001	.0001	--	--	--	--
Pu-239	--	--	--	.0044	--	.0056	.9025	.2277	.3054	.0324
Pu-240	--	--	--	--	--	--	.0026	.0025	.0016	.0010
Pu-241	--	--	--	0.0014	--	.0016	.2960	.3128	.2302	.1592
Pu-242	--	--	--	--	--	--	0.0004	0.0008	0.0018	0.0019
Total	1.1584	0.9479	1.1363	0.9499	1.1437	0.9494	1.2085	0.9194	1.1566	0.9412

Table 2-2. (Cont'd)

	<u>Inner tube (in-core section)</u>	<u>Outer tube (in-core section)</u>	<u>Lower fuel channel support tube</u>	<u>Upper fuel channel support tube</u>
Wall thickness, in.	0.122	0.059	0.25	0.25
Length, ft-in.	20-0	20-0	3-7	3-7

Overall fuel channel assembly length, ft-in.	37-2 (414 PT's) 38-8 (548 PT's)
Insulation between inner and outer tubes	Carbon dioxide
Height of fuel channel assembly extending above top shield, ft-in.	5-10 (414 PT's) 6-7 (548 PT's)
Height of fuel channel assembly extending below bottom shield, ft	5-4 (414 PT's) 6-1 (548 PT's)
Direction of coolant flow in fuel channels	Up & down
Direction of fuel flow in fuel channels	Up & down
Method of refueling	On-line, by machine, from top & bottom

Moderator Tank

Material	SS
ID, ft-in.	23-1 (414 PT's) 26-3 (548 PT's)
Wall thickness, in.	1
Inside shield length, ft-in.	20-0
Top shield height, ft-in.	3-0
Top shield diameter, ft-in.	25-1 (414 PT's) 28-3 (548 PT's)
Bottom shield height, ft	3
Bottom shield diameter, ft-in.	25-1 (414 PT's) 28-3 (548 PT's)

Piping (for 548 PT's)

Total number of pipes from fuel channel assemblies to header (each end)	548
Piping material	Carbon steel

Table 2-2. (Cont'd)

II. Fuel Element Data (for 414 PT's)

General

Figure number	2-2
Fuel assemblies/fuel channel	5
Fuel assemblies/reactor	2070
Fuel cylinders/assembly	5
Refueling rate, assemblies/day @ 20,000 MWD/T	2.5
Fuel content/assembly, kg (U + Th)	52.1
Fuel element weight, lb	140
Materials composition inside homogenized fuel channel, vol %	
Fuel	42.6
Structural material (all Zr)	22.1
Coolant	35.3

Dimensions (Hot)

Active fuel length/assembly, in.	42.66
Overall length (effective), in.	43.96
Zr-2 center flow blocker diameter, in.	0.280
Zr-4 cladding thickness, in. (typical for all cylinders)	0.025
Fuel thickness, in.	
Cyl 1	0.284
Cyl 2	0.240
Cyl 3	0.175
Cyl 4	0.126
Cyl 5	0.090
Coolant gap thickness, in.	
Flow blocker - cyl 1	0.11
Cyl 1 - 2	0.125
Cyl 2 - 3	0.125
Cyl 3 - 4	0.125
Cyl 4 - 5	0.125
Cyl 5 - pressure tube	0.100

Table 2-2. (Cont'd)

III. Thermal and Hydraulic Data (for 414 PT's)

General

Core inlet temperature, F	560
Core mixed mean outlet temperature, F	711
Total coolant flow, lb/hr	119×10^6
Core inlet pressure, psi	383
Total core pressure drop (including header), psi	300
Minimum ratio, critical flux to actual flux	2.2

Core

General

Number of pressure tubes	414
Power to coolant, MWt	3039
Pressure drop, active core including head, psi	246
Pressure drop, headers, psi	54
Fission power/avg pressure tube, MWt	7.755
Cylinders/element	5

Fuel Tubes

Nominal Pressure Tube^(a)

Power/pressure tube, MWt	8.59
Mixed mean outlet temperature, F	711

	<u>Cylinder</u>				
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>
Max surface temp, F					
ID	769	771	771	775	772
OD	765	769	772	775	756
Max heat flux, Btu/hr-ft ² × 10 ⁻⁵					
ID	3.63	3.44	3.38	3.47	3.57
OD	3.20	3.31	3.34	3.45	3.55

(a) Nominal pressure tube has nominal dimensions and nominal power peaking factors.

Table 2-2. (Cont'd)

	Cylinder				
	1	2	3	4	5
Max fuel temp, F	884	880	867	862	845
Peak-to-average power ratio	0.523	0.623	0.854	1.220	1.759

Hot Pressure Tube^(b)

Power/pressure tube, MWt	9.32
Mixed mean outlet temp, F	737

	Cylinder				
	1	2	3	4	5
Max surface temp, F					
ID	782	784	784	795	819
OD	779	782	787	813	803
Max heat flux, Btu/hr-ft ² × 10 ⁻⁵					
ID	3.87	3.67	3.61	3.75	4.0
OD	3.41	3.53	3.56	3.61	3.95
Max fuel temp, F	908	901	887	900	905
Peak-to-average power ratio	0.541	0.644	0.882	1.139	1.662

Coolant Channels

	Channel					
	1	2	3	4	5	6
Nominal Pressure Tube						
Channel thickness, in.	0.110	0.125	0.125	0.125	0.125	0.100
Channel flow, lb/hr × 10 ⁻³	9.4	35.0	56.7	24.9	90.6	72.7

(b) Hot pressure tube has maximum power peaking factors and worst tolerances for geometry and power generation. Dimensional variations caused by fabrication tolerances and fuel growth are incorporated in the fuel element geometry used in the thermal analysis. The outer cylinder is assumed to have a power generation factor of 1.05.

Table 2-2. (Cont'd)

	Channel					
	1	2	3	4	5	6
Avg velocity, fps	45.5	49.2	49.2	49.2	49.3	42.6
Channel exit temp, F	703	715	717	719	720	680
 Hot Pressure Tube						
Channel thickness, in.	0.11	0.125	0.125	0.125	0.112	0.087
Channel flow, lb/hr $\times 10^{-3}$	9.4	35.0	56.6	74.8	76.2	58.3
Avg velocity, fps	45.5	49.3	49.3	49.4	46.7	39.6
Channel exit temp, F	712	726	727	733	763	725

IV. Nuclear Data

Idealized U-233 Cycle

U-233, wt %	1.85	1.90
k_{eff}	1.1541	1.1671
Days	556	646
Tons	0.24660	0.24670
MWD/T	17,476	20,296
C. R.	0.9132	0.9061
Watts/gram	31.43	31.43
Fuel fraction	0.436	0.436

Masses

Time 0

Th	241.789	241.757
U-233	4.562	4.687
U-234	0.252	0.257

Final

Th	237.389	236.695
U-233	3.732	3.736
U-234	0.638	0.692
U-235	0.090	0.104
U-236	0.010	0.014
Pa	0.319	0.318

$P/\bar{P}$ Radial

Cyl 1	0.521
2	0.622
3	0.852
4	1.219
5	1.757

Table 2-3. Core Design Data for 37-Pin ThO₂ Fuel Element
in a 4.030-Inch-ID Pressure Tube

I. Reactor Data

General

Moderator	D ₂ O
Coolant	Santowax OM-10% HB
Fuel	ThO ₂ + UO ₂ Vipac pins
Fission power, MWt	3100
Thermal power to coolant, MWt	2936
Thermal power to moderator, MWt	157
Net electric power, MWe	1076
Number of coolant passes	1
Active fuel length, ft-in. (5 × active fuel length/element)	17-9.4
Active core length, ft-in. (fuel to fuel for 5 elements)	18-6.0
Overall stack length, ft-in. (end plate to end plate)	18-8.1
Peak-to-average power ratios	
Gross radial	1.17
Local radial, effective	1.25
Gross axial	1.3
Local axial	1.17
Flux tilt	1.05
Pitch of fuel channel assemblies, in.	9.5
D ₂ O inventory in calandria, lb	624,854
D ₂ O cover gas	Helium

Fuel Channel Assembly

	Inner tube (in-core section)	Outer tube (in-core section)	Lower fuel channel support tube	Upper fuel channel support tube
Material	SAP	Zr-2	SS	SS
OD, in.	4.248	5.127	6.43	6.43
Wall thickness, in.	0.109	0.069	0.25	0.25

Table 2-3. (Cont'd)

	<u>Inner tube (in-core section)</u>	<u>Outer tube (in-core section)</u>	<u>Lower fuel channel support tube</u>	<u>Upper fuel channel support tube</u>
Length, ft-in.	20-4	20-4	3-7	3-7
Overall fuel channel assembly length, ft-in.				40-11.2
Insulation between inner and outer tubes				Carbon dioxide
Height of fuel channel assembly extending above top shield, ft-in.				7-6.6
Height of fuel channel assembly extending below bottom shield, ft-in.				7-0.6
Total number of fuel channels in core				758
Direction of coolant flow				Up & down
Direction of fuel flow				Up & down
Method of refueling				On-line, from top & bottom
How fuel channel is isolated				Manually operated valves
<u>Moderator Tank</u>				
Material				304 SS
ID, ft-in.				26-7
Wall thickness, in.				1
Inside shield length, ft-in.				20-4
Top shield height, ft-in.				3-0
Top shield diameter, ft-in.				28-7
Bottom shield height, ft-in.				3-0
Bottom shield diameter, ft-in.				28-7
<u>Piping</u>				
Total number of pipes from fuel channel assemblies to header (each end)				758
Joint between fuel channel assemblies & pipes to headers				Mechanical
Piping material				Carbon steel

Table 2-3. (Cont'd)

II. Fuel Element Data

General

Drawing number A30471 E	Figure 2-13
Fuel elements/fuel channel	5
Fuel elements/core	3790
Fuel density, % of theoretical	88
Fuel element assembly weight, lb	88
U + Th/fuel element assembly, kg	31.6
Refueling rate, elements/day @ 20,000 MWD/T (avg)	3.9
Composition of homogenized fuel channel, vol %	
Fuel	43.46
Santowax	40.75
SAP	14.13
Zircaloy	0.53
Void	1.13

Individual Fuel Pins (Hot Dimensions)

Clad material	SAP 895
Clad OD, in.	
Large pin/small pin	0.515/0.332
Cladding thickness, in.	
Large pin/small pin	0.022/0.017
U + Th/fuel rod, kg (31.6/37)	0.855
Fuel rods/element	
Large pin/small pin	31/6
Fuel rods/core	
Large pin/small pin	117,490/22,740
Fins/fuel rod (spiral)	
Large pin/small pin	12/6
Fin revolutions/ft of length	
Large pin/small pin	90°/90°
Height of fins, in.	
Large pin/small pin	0.058/0.038, 0.102
Width of fins, in.	0.030
Surface finish	63
Fission gas gap, in.	
Large pin/small pin	1.30/0.430

Table 2-3. (Cont'd)

Dimensions (Hot)

Pitch between fuel rods, in. (large pin)	0.631
Fuel element diameter, in.	
Mid-section/ends	3.970/3.983
Diametral clearance between element & inner tube (in-core), in.	
Mid-section/ends	0.060/0.047
Fuel element length	
Overall, in.	45.17
Overend plates, in.	44.82
Active fuel length (equivalent) per fuel element, in.	42.66

III. Thermal and Hydraulic Data

General

Core inlet temp, F	560
Core outlet temp, F	721.4
Outlet temp, hottest pressure tube, F	721.4
Coolant flow, lb/hr $\times 10^{-6}$	107.2
Core inlet pressure, psia	229
Pressure drop (including headers), psi	146
Avg linear heat rate (hot pin), kw/ft	10
Minimum burnout ratio	2.59

Core

General

Power to coolant, MWt	2936
Pressure drop, psi	
Active core (excluding head)	92
Control valve and headers	54
Power/hot pressure tube, MWt	4.53
Avg fission power/pressure tube, MWt	4.09
Avg coolant flow/pressure tube, lb/hr $\times 10^{-5}$	1.41
Pressure Tube (Maximum Power)	
Power, MWt	4.53
Coolant flow, lb/hr $\times 10^{-5}$	1.65

Table 2-3. (Cont'd)

Coolant outlet temp, F		721.4
Pressure drop, psi		
Active core		92
Headers		54
	<u>Hot</u> <u>condition</u>	<u>Nominal</u> <u>condition</u>
Unit Cell		
Coolant outlet temp, F	762	746
Max surface temp, F	830	804
Max heat flux, Btu/hr-ft ² × 10 ⁻⁵	2.23	2.09
Surface area (including fins), ft ²	4.26	4.26
Max linear heat rate (hot spot), kw/ft	15.66	14.68
Coolant velocity, avg fps	21.7	21.6
Hot channel factors		
Coolant temp rise, F ΔT	1.09	1.0
Film temp drop, F θ	1.16	1.0
Heat flux, F q''	1.07	1.0
 <u>IV. Nuclear Data</u>		
<u>Fuel Loading/Fuel Channel (U + Th Mctal), kg</u>		158.3
<u>Fuel Volume Fraction of Homogenized Element Within Liner</u>		0.435
 <u>U-233 Idealized Cycle</u>		
Burnup, MWD/tonne		21,924
Fuel enrichment, wt % U-233		1.86
Conversion ratio		0.914
BOL mass balances, kg/pressure tube		
U-233		2,945
U-234		0.173
Th-232		<u>154.840</u>
Total		157.958

Table 2-3. (Cont'd)

EOL mass balances, kg/pressure tube	
U-233	2.353
U-234	0.461
U-235	0.074
U-236	0.010
Pa-233	0.183
Th-232	<u>151.312</u>
Total	154.393

Ratio of Power to Average Power ($P/\bar{P}$) at Time Zero

Local radial $P/\bar{P}$	
Pin location (Figure 2-3)	$P/\bar{P}$ idealized cycle
1	0.621
2	0.680
3	0.906
4	1.412
5	0.680
6	0.818
7	1.248
8	0.818
9	0.906
10	1.248
11	1.248
12	1.412
Local axial $P/\bar{P}$ (BAW-393-6)	1.17
Gross radial $P/\bar{P}$ (section 2.3.1.3)	1.17
Gross axial $P/\bar{P}$ (section 2.3.1.3)	1.29

Table 2-3. (Cont'd)

Mass Balances, U-235 Startup and Makeup

Isotope	Mass balance per pressure tube, kg											
	Cycle 1		Cycle 2		Cycle 3		Cycle 4		Cycle 5		Cycle 6	
	BOL	EOL	BOL	EOL	BOL	EOL	BOL	EOL	BOL	EOL	BOL	EOL
U-233	--	1.542	1.674	2.038	2.166	2.175	2.300	2.216	2.339	2.228	2.348	2.231
U-234	--	0.143	0.141	0.342	0.337	0.509	0.502	0.645	0.635	0.746	0.735	0.822
U-235	3.656	1.075	1.621	0.492	1.039	0.371	0.921	0.351	0.893	0.368	0.902	0.384
U-236	--	0.415	0.408	0.571	0.562	0.650	0.640	0.715	0.704	0.772	0.759	0.827
U-238	0.279	0.260	0.298	0.277	0.314	0.293	0.331	0.307	0.342	0.320	0.355	0.332
Pa-233	--	0.159	--	0.163	--	0.162	--	0.161	--	0.159	--	0.158
Pu-239, Pu-241	--	0.006	--	0.006	--	0.007	--	0.007	--	0.007	--	0.007
Th-232	154.378	151.398	154.166	151.064	153.895	150.878	153.617	150.507	153.395	150.372	153.208	150.205
Total	158.313		158.303		158.313		158.311		158.308		158.307	
<u>Cycle Conditions</u>												
E, wt % U-233	--		1.06		1.37		1.45		1.48		1.48	
E, wt % U-235	2.31		1.02		0.66		0.58		0.56		0.57	
Power, MW		4.09		4.09		4.09		4.09		4.09		4.09
Full power days		780		786		764		794		778		778
MWD/tonne		20,152		20,308		19,738		20,513		20,100		20,100
Conversion ratio		0.7770		0.8433		0.8654		0.8709		0.8730		0.8733
<u>Recycle Masses After 180-Day Decay and 1.5% Reprocessing Losses Plus Makeup Mass, kg</u>												
Isotope	Recycle	Makeup	Recycle	Makeup	Recycle	Makeup	Recycle	Makeup	Recycle	Makeup	Recycle	Makeup
U-233	1.674	--	2.166	--	2.300	--	2.339	--	2.348	--		
U-234	0.141	--	0.337	--	0.502	--	0.635	--	0.735	--		
U-235	1.059	0.562	0.485	0.554	0.365	0.556	0.346	0.547	0.362	0.540		
U-236	0.408	--	0.562	--	0.640	--	0.704	--	0.759	--		
U-238	0.256	0.042	0.273	0.041	0.285	0.042	0.302	0.040	0.315	0.040		
Th-232	149.127	5.039	148.798	5.097	148.615	5.002	148.249	5.146	148.116	5.092		

Table 2-3. (Cont'd)

Mass Balances, Plutonium Startup and Makeup

Isotope	Mass balance per pressure tube, kg											
	Cycle 1		Cycle 2		Cycle 3		Cycle 4		Cycle 5		Cycle 6	
	BOL	EOL	BOL	EOL	BOL	EOL	BOL	EOL	BOL	EOL	BOL	EOL
U-233	--	1.335	1.446	1.976	2.077	2.232	2.330	2.314	2.414	2.335	2.438	2.335
U-234	--	0.097	0.095	0.241	0.237	0.391	0.385	0.523	0.516	0.626	0.616	0.722
U-235	--	0.010	0.010	0.039	0.038	0.075	0.075	0.109	0.107	0.134	0.132	0.158
U-236	--	0.001	0.001	0.005	0.005	0.014	0.014	0.029	0.028	0.046	0.045	0.070
Pu-239	3.078	0.500	1.420	0.176	0.859	0.081	0.664	0.054	0.602	0.052	0.611	0.036
Pu-240	1.679	1.485	1.969	1.323	1.678	0.999	1.301	0.742	1.030	0.606	0.902	0.484
Pu-241	0.741	0.522	0.737	0.445	0.603	0.338	0.473	0.257	0.385	0.216	0.347	0.176
Pu-242	0.333	0.599	0.690	0.929	0.988	1.153	1.200	1.287	1.327	1.362	1.403	1.406
Pa-233	--	0.133	--	0.133	--	0.136	--	0.139	--	0.140	--	0.144
Th-232	152.779	150.546	152.260	149.914	152.146	149.631	152.185	149.622	152.218	149.772	152.136	149.308
Total	158.630		158.678		158.631		158.627		158.627		158.630	

Cycle Conditions

E, wt % U-233	--		0.91		1.31		1.47		1.52		1.54	
E, wt % Pu-239	1.94		0.90		0.54		0.42		0.38		0.39	
E, wt % Pu-241	0.47		0.46		0.38		0.30		0.24		0.22	
Power, MW		4.09		4.09		4.09		4.09		4.09		4.09
Full power days		790		786		810		800		750		752
MWD/tonne		20,369		20,266		20,884		20,627		19,337		19,389
Conversion ratio		0.7278		0.8100		0.8357		0.8404		0.8371		0.8448

Recycle Masses After 180-Day Decay and 1.5% Reprocessing Losses Plus Makeup Masses, kg

Isotope	Recycle	Makeup	Recycle	Makeup	Recycle	Makeup	Recycle	Makeup	Recycle	Makeup
U-233	1.446	--	2.077	--	2.330	--	2.414	--	2.438	--
U-234	0.095	--	0.237	--	0.385	--	0.516	--	0.616	--
U-235	0.010	--	0.038	--	0.075	--	0.107	--	0.132	--
U-236	0.001	--	0.005	--	0.014	--	0.028	--	0.045	--
Pu-239	0.492	0.928	0.172	0.687	0.080	0.584	0.053	0.549	0.051	0.560
Pu-240	1.463	0.506	1.304	0.374	0.983	0.318	0.731	0.299	0.597	0.305
Pu-241	0.514	0.223	0.438	0.165	0.333	0.140	0.253	0.132	0.212	0.135
Pu-242	0.590	0.100	0.914	0.074	1.137	0.063	1.268	0.059	1.343	0.060
Th-232	148.288	3.972	147.665	4.481	147.386	4.799	147.378	4.840	147.525	4.611

Table 2-4. Core Design Data for 55-Pin Thorium Oxide Fuel Element in a 4.030-Inch-ID Pressure Tube

I. Reactor Data

General

Moderator	D ₂ O
Coolant	Santowax OM-10% HB
Fuel	ThO ₂ + UO ₂ Vipac pins
Fission power, MWt	3183
Thermal power to coolant, MWt	3015
Thermal power to moderator, MWt	158
Net electric power, MWe	1076
Number of coolant passes	1
Active fuel length, ft-in. (5 × active fuel length/element)	17-6
Active core length, ft-in. (fuel to fuel for 5 elements)	17-10
Overall stack length, ft-in. (end plate to end plate)	18-4
Peak-to-average power ratios	
Gross radial	1.17
Local radial	1.37
Gross axial	1.33
Local axial	1.17
Flux tilt	1.05
Pitch of fuel channel assemblies, in.	11.0
D ₂ O inventory in calandria, lb	642,768
D ₂ O cover gas	Helium

Fuel Channel Assembly

	Inner tube (in-core section)	Outer tube (in-core section)	Lower fuel channel support tube	Upper fuel channel support tube
Material	SAP	Zr-2	SS	SS
OD, in.	4.274	4.992	6.35	6.35
Wall thickness, in.	0.122	0.059	0.25	0.25

Table 2-4. (Cont'd)

	Inner tube (in-core section)	Outer tube (in-core section)	Lower fuel channel support tube	Upper fuel channel support tube
Length, ft-in.	20-0	20-0	3-7	3-7
Overall fuel channel assembly length, ft-in.				38-8
Insulation between inner and outer tubes			Carbon dioxide	
Height of fuel channel assembly extending above top shield, ft-in.				6-7
Height of fuel channel assembly extending below bottom shield, ft-in.				6-1
Total number of fuel channels in core				548
Direction of coolant flow			Up & down	
Direction of fuel flow			Up & down	
Method of refueling			On-line, from top & bottom	
<u>Moderator Tank</u>				
Material				304 SS
ID, ft-in.				26-3
Wall thickness, in.				1
Inside shield length, ft-in.				20-0
Top shield height, ft-in.				3-0
Top shield diameter, ft-in.				28-3
Bottom shield height, ft-in.				3-0
Bottom shield diameter, ft-in.				28-3
<u>Piping</u>				
Total number of pipes from fuel channel assemblies to header (each end)				548
Piping material			Carbon steel	

Table 2-4. (Cont'd)

II. Fuel Element Data

General

Figure number	Figure 2-20
Fuel elements/fuel channel	5
Fuel elements/core	2740
Fuel density, % of theoretical	88
Fuel element assembly weight, lb	~70
(U + Th)/fuel element assembly, kg	25.2
Refueling rate, elements/day @ 20,000 MWD/T, avg	5.1
Composition of homogenized fuel channel, vol %	
Fuel	35.28
Santowax	43.24
SAP	19.90
Zircaloy	0.49
Void	1.05
Inconel-X	0.04

Individual Fuel Pins (Hot Dimensions)

Clad material	SAP 895
Clad OD, in.	
Large pin/small pin	0.381/0.329
Clad wall thickness, in. (large & small pins)	0.020
(U + Th)/fuel rod, kg (25.2/55)	0.458
Fuel rods/element	
Large pin/small pin	43/12
Fuel rods/core	
Large pin/small pin	117,820/32,880
Spiral fins per fuel rod (large & small)	6
Fin revolutions/ft length (large & small)	0.83
Fin height, in.	
Large pin/small pin	0.060/0.050
Fin width, in.	0.030
Surface finish	63
Fission gas gap, in.	
Large pin/small pin	1.300/0.930

Table 2-4. (Cont'd)

Dimensions (Hot)

Pitch between fuel rods, in. (large pin)	0.501
Fuel element diameter, in.	
Mid-section/ends	3.970/3.988
Diametral clearance between fuel element and inner tube (in-core section), in.	
Mid-section/ends	0.060/0.042
Fuel element length, in.	
Overall/over end plates	44.350/44.000
Active fuel length (equivalent)/fuel element, in.	42.000

III. Thermal and Hydraulic Data

General

Core inlet temp, F	560
Core outlet temp, F	706.5
Hottest pressure tube outlet temp, F	706.5
Coolant flow, lb/hr $\times 10^{-6}$	121.7
Inlet pressure, psia	310
Pressure drop (including headers), psi	227
Avg hot pin linear heat rate, kw/ft	10.24
Minimum burnout ratio	2.25

Core

General

Power to coolant, MWt	3015
Pressure drop, psi	
Active core, excluding head	173
Headers	54
Max power/hot pressure tube, MWt	6.43
Avg fission power/pressure tube, MWt	5.81
Avg coolant flow/pressure tube, lb/hr $\times 10^{-5}$	2.22
Pressure Tube (Maximum Power)	
Power, MWt	6.43
Coolant flow, lb/hr $\times 10^{-5}$	2.60
Coolant outlet temp, F	706.5

Table 2-4. (Cont'd)

Pressure drop, psi		
Active core		173
Headers		54
	<u>Hot condition</u>	<u>Nominal condition</u>
Unit Cell		
Coolant outlet temp, F	734.4	720.4
Max surface temp, F	830	802.9
Max heat flux, Btu/hr-ft ² × 10 ⁻⁵	3.88	3.63
Surface area (including fins), ft ²	2.47	2.47
Max linear heat rate (hot spot), kw/ft	16.06	15.01
Avg coolant velocity, fps	35.5	35.4
Hot channel factors		
Coolant temp rise, F _{ΔT}	1.09	1.0
Film temp drop, F _θ	1.16	1.0
Heat flux, F _{q''}	1.07	1.0

IV. Nuclear Data

Fuel Loading/Fuel Channel (U + Th Metal), kg	126.0
Fuel Volume Fraction of Homogenized Element Within Liner	0.353
<u>U-233 Idealized Cycle</u>	
Burnup, MWD/tonne	19,291
Fuel enrichment, wt % U-233	2.04
Conversion ratio	0.853
BOL mass balances, kg/pressure tube	
U-233	2.574
U-234	0.150
Th-232	123.340
Total	126.064

Table 2-4. (Cont'd)

EOL mass balances, kg/pressure tube

U-233	1.904
U-234	0.360
U-235	0.047
U-236	0.006
Pa-233	0.224
Th-232	<u>121.048</u>
Total	123.589

Ratio of Local Radial Power to Average Power ($P/\bar{P}$)
at Time Zero

<u>Pin location (Figure 2-4)</u>	<u>$P/\bar{P}$ idealized cycle</u>
1	0.602
2	0.636
3	0.768
4	1.071
5	0.636
6	0.723
7	0.936
8	1.436
9	0.723
10	0.768
11	0.936
12	1.369
13	0.936
14	1.071
15	1.436
16	1.369
17	1.436

Table 2-5. Core Design Data for 37-Pin Thorium Carbide
Fuel Element in a 4.030-Inch-ID Pressure Tube

I. Reactor Data

General

Moderator	D ₂ O
Coolant	Santowax OM-10% HB
Fuel	ThC + UC slugs
Fission power, MWt	3151
Thermal power to coolant, MWt	2984
Thermal power to moderator, MWt	157
Net electric power, MWe	1076
Number of coolant passes	1
Active fuel length, ft-in. (5 × active fuel length/element)	17-9.4
Active core length, ft-in. (fuel to fuel for 5 elements)	18-3.2
Overall stack length, ft-in. (end plate to end plate)	18-4.6
Peak-to-average power ratios	
Gross radial	1.17
Local radial, effective	1.28
Gross axial	1.33
Local axial	1.17
Flux tilt	1.05
Pitch of fuel channel assemblies, in.	11.0
D ₂ O inventory in calandria, lb	642,768
D ₂ O cover gas	Helium

Fuel Channel Assembly

	Inner tube (in-core section)	Outer tube (in-core section)	Lower fuel channel support tube	Upper fuel channel support tube
Material	SAP	Zr-2	SS	SS
OD, in.	4.274	4.992	6.35	6.35
Wall thickness, in.	0.122	0.059	0.25	0.25

Table 2-5. (Cont'd)

	Inner tube (in-core section)	Outer tube (in-core section)	Lower fuel channel support tube	Upper fuel channel support tube
Length, ft-in.	20-0	20-0	3-7	3-7
Overall fuel channel assembly length, ft-in.				38-8
Insulation between inner and outer tubes				Carbon dioxide
Height of fuel channel assembly extending above top shield, ft-in. {				6-7
Height of fuel channel assembly extending below bottom shield, ft-in.				6-1
Total number of fuel channels in core				548
Direction of coolant flow				Up & down
Direction of fuel flow				Up & down
Method of refueling				On-line, from top and bottom
<u>Moderator Tank</u>				
Material				304 SS
ID, ft-in.				26-3
Wall thickness, in.				1
Inside shield length, ft-in.				20-0
Top shield height, ft-in.				3-0
Top shield diameter, ft-in.				28-3
Bottom shield height, ft-in.				3-0
Bottom shield diameter, ft-in.				28-3
<u>Piping</u>				
Total number of pipes from fuel channel assemblies to header (each end)				548
Piping material				Carbon steel

Table 2-5. (Cont'd)

II. Fuel Element Data

General

Drawing number	Figure 2-23
Fuel elements/fuel channel	5
Fuel elements per core	2740
Fuel density (including dish & gap), % of theoretical	94.26
Fuel element assembly weight, lb	102
U + Th/fuel element assembly, kg	36.12
Refueling rate, elements/day @ 20,000 MWD/T (avg)	3.5
Composition of homogenized fuel channel, vol %	
Fuel	40.71
Santowax	43.48
SAP	14.76
Zircaloy	0.56
Void	0.49

Individual Fuel Pins (Hot Dimensions)

Clad material	SAP 895
Clad OD, in.	
Large pin/small pin	0.495/0.332
Cladding wall thickness, in.	
Large pin/small pin	0.022/0.017
U + Th/fuel rod, kg (36.12/37)	0.976
Fuel rods/element	
Large pin/small pin	31/6
Fuel rods/core	
Large pin/small pin	84,940/16,440
Fins/fuel rod (spiral)	
Large pin/small pin	12/6
Fin revolutions/ft length	
Large pin/small pin	90°/90°
Height of fins, in.	
Large pin/small pin	0.068/0.038, 0.102
Width of fins, in.	0.030
Surface finish	63
Fission gas gap, in.	
Large pin/small pin	0.600/0.200

Table 2-5. (Cont'd)

Dimensions (Hot)

Pitch between fuel rods, in. (large pin)	0.631
Fuel element diameter, in. Mid-section/ends	3.970/3.983
Diametral clearance between fuel element and inner tube (in-core section), in. Mid-section/ends	0.060/0.047
Fuel element length, in. Overall/over end plates	44.47/44.12
Active fuel length (equivalent)/fuel element, in.	42.66

III. Thermal and Hydraulic Data

General

Core inlet temp, F	560
Core outlet temp, F	716.2
Outlet temp, hottest pressure tube, F	716.2
Coolant flow, lb/hr $\times 10^{-6}$	113
Core inlet pressure, psia	292
Pressure drop (including headers), psi	209
Avg linear heat rate (hot pin), kw/ft	14.3
Minimum burnout ratio	2.34

Core

General

Power to coolant, MWt	2984
Pressure drop, psi	
Active core (excluding head)	155
Headers	54
Power/hot pressure tube, MWt	6.37
Avg fission power/pressure tube, MWt	5.75
Avg coolant flow/pressure tube, lb/hr $\times 10^{-5}$	2.06
Pressure Tube (Maximum Power)	
Power, MWt	6.37
Coolant flow, lb/hr $\times 10^{-5}$	2.41
Coolant outlet temp, F	716.2

Table 2-5. (Cont'd)

Pressure drop, psi		
Active core		155
Headers		54
	<u>Hot</u>	<u>Nominal</u>
	<u>condition</u>	<u>condition</u>
Unit Cell		
Coolant outlet temp, F	751.2	735.9
Max surface temp, F	830	803.6
Max heat flux, Btu/hr-ft ² × 10 ⁻⁵	3.13	2.92
Surface area (including fins), ft ²	4.35	4.35
Max linear heat rate (hot spot), kw/ft	22.41	20.94
Coolant velocity, avg fps	30.7	30.6
Hot channel factors		
Coolant temp rise, F ΔT	1.09	1.0
Film temp drop, F θ	1.16	1.0
Heat flux, F q''	1.07	1.0
 <u>IV. Nuclear Data</u>		
<u>Fuel Loading/Fuel Channel</u> <u>(U + Th Metal), kg</u>		180.6
<u>Fuel Volume Fraction of Homogenized Element</u> <u>Within Liner</u>		0.407
 <u>U-233 Idealized Cycle</u>		
Burnup, MWD/tonne		21,997
Fuel enrichment, wt % U-233		1.87
Conversion ratio		0.906
BOL mass balances, kg/pressure tube		
U-233		3.376
U-234		0.197
Th-232		176.775
Total		180.348

Table 2-5. (Cont'd)

EOL mass balances, kg/pressure tube	
U-233	2.647
U-234	0.527
U-235	0.080
U-236	0.011
Pa-233	0.238
Th-232	172.768
Total	176.271

Ratio of Power to Average Power ($P/\bar{P}$) at Time Zero

<u>Local radial $P/\bar{P}$</u>		
<u>Pin location</u> <u>(Figure 2-3)</u>	<u>Idealized cycle</u>	<u>U-235 cycle 1</u>
1	0.555	0.486
2	0.616	0.555
3	0.880	0.846
4	1.487	1.593
5	0.616	0.555
6	0.780	0.729
7	1.293	1.348
8	0.780	0.729
9	0.880	0.846
10	1.293	1.348
11	1.293	1.348
12	1.487	1.593
Local axial $P/\bar{P}$ (BAW-393-6)		1.17
Gross radial $P/\bar{P}$ (section 2.3.1.3)		1.17
Gross axial $P/\bar{P}$ (section 2.3.1.3)		1.30

Table 2-5. (Cont'd)

Mass Balances, U-235 Startup and Makeup

Isotope	Mass balance per pressure tube, kg											
	Cycle 1		Cycle 2		Cycle 3		Cycle 4		Cycle 5		Cycle 6	
	BOL	EOL	BOL	EOL	BOL	EOL	BOL	EOL	BOL	EOL	BOL	EOL
U-233	--	1.703	1.895	2.259	2.447	2.420	2.605	2.466	2.649	2.481	2.662	2.484
U-234	--	0.164	0.162	0.387	0.381	0.582	0.573	0.739	0.728	0.858	0.844	0.950
U-235	4.200	1.226	1.898	0.578	1.259	0.432	1.115	0.412	1.088	0.430	1.098	0.441
U-236	--	0.472	0.464	0.656	0.647	0.761	0.749	0.845	0.831	0.922	0.908	0.998
U-238	0.320	0.301	0.348	0.325	0.373	0.350	0.396	0.371	0.415	0.390	0.434	0.407
Pa-233	--	0.224	--	0.223	--	0.227	--	0.225	--	0.223	--	0.221
Pu-239, Pu-241	--	0.699	--	0.006	--	0.007	--	0.007	--	0.007	--	0.008
Th-232	176.062	172.699	175.740	172.307	175.498	172.050	175.096	171.600	174.935	171.523	174.613	171.147
Total	180.582		180.507		180.595		180.534		180.646		180.559	

Cycle Conditions

E, wt % U-233	--		1.05		1.35		1.44		1.47		1.47	
E, wt % U-235	2.32		1.05		0.70		0.62		0.60		0.61	
Power, MW		5.8		5.8		5.8		5.8		5.8		5.8
Full power days		632		624		624		640		630		644
MWD/tonne		20,299		20,050		20,040		20,562		20,227		20,687
Conversion ratio		0.7663		0.3303		0.8524		0.8578		0.8591		0.8600

Recycle Masses After 180-Day Decay and 1.5% Reprocessing Losses Plus Makeup, kg

Isotope	Recycle	Makeup	Recycle	Makeup	Recycle	Makeup	Recycle	Makeup	Recycle	Makeup
U-233	1.895	--	2.447	--	2.605	--	2.649	--	2.662	--
U-234	0.162	--	0.381	--	0.573	--	0.728	--	0.844	--
U-235	1.209	0.689	0.570	0.689	0.425	0.689	0.406	0.682	0.423	0.675
U-236	0.464	--	0.647	--	0.749	--	0.831	--	0.908	--
U-238	0.296	0.052	0.321	0.052	0.344	0.052	0.364	0.051	0.383	0.051
Th-232	170.034	5.706	169.734	5.754	169.423	5.673	169.092	5.843	168.929	5.684

Figure 2-1. HWOCR 4-Cylinder Thorium Metal Fuel Element (With Center Moderator) (A30461 E)

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Figure 2-2. Bow Condition Between Two Adjacent Cylinders

Figure 2-3. Gross Axial Power Distribution

Figure 2-4. Radial Power Distribution
(Three Equal Volume Zones)

Figure 2-5. Axial Local Power Peaking ($P/\bar{P}$) in Nested-Cylinder Fuel Element

U-233 Idealized Cycle

U-235 Startup Cycle

Note: All dimensions in cm.

Figure 2-6. Coolant, Cladding, and Fuel Temperature Vs Active Fuel Length (Hottest Cylinder and Coolant Channel — Nominal Conditions)

Figure 2-7. Coolant, Cladding, and Fuel Temperature Vs Active Fuel Length (Hottest Cylinder and Coolant Channel — Hot Channel Factors)

Figure 2-8. Ratio of Burnout Heat Flux to Actual Heat Flux,
Based on Constant Saturation Temperature of
850 F

Figure 2-9. Five-Cylinder Thorium Metal Fuel Element,
Cross Section

Figure 2-10. Coolant, Cladding, and Fuel Temperatures Vs Active Fuel Length (Hottest Cylinder and Coolant Channel — Nominal Conditions)

Figure 2-11. Coolant, Cladding, and Fuel Temperatures Vs Active Fuel Length (Hottest Cylinder and Coolant Channel — Hot Channel Factors)

Figure 2-12. Ratio of Burnout Heat Flux to Actual Heat Flux,
Based on Constant Saturation Temperature of
850 F

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Figure 2-13. HWO CR 37-Pin ThO_2
Fuel Element
(A30471 E)

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Figure 2-14. Axial Local Power Peaking ($P/\bar{P}$) in Clustered-Pin Fuel Element

U-233 Idealized Cycle

Note: All dimensions in cm.

Figure 2-15. Flow-Channel Designation, 37-Pin ThO_2 Fuel Element

Figure 2-16. Hot Channel Coolant and Cladding Surface Temperature Vs Active Fuel Length

Figure 2-17. Hot Fuel Pin Temperature Vs Active Fuel Length

Figure 2-18. Ratio of Burnout Heat Flux to Actual Flux,
Based on Constant Saturation Temperature
of 850 F

Figure 2-19. Fission Gas Pressure in Hot Pin Vs Full Power Days

Figure 2-20. 55-Pin ThO_2 Fuel Element, Cross Section

Figure 2-21. Coolant and Cladding Temperatures Vs Active Fuel Length, 55-Pin ThO_2 Fuel Element in 4.030-Inch-ID Pressure Tube

Figure 2-22. Ratio of Burnout Flux to Actual Flux, Based on Constant Saturation Temperature of 850 F

Figure 2-23. Flow-Channel Designation, 37-Pin Thorium Carbide Fuel Element

Figure 2-24. Coolant and Cladding Temperatures Vs Active Fuel Length, Thorium Carbide Fuel Element

Figure 2-25. Ratio of Burnout Flux to Actual Flux, Based on Constant Saturation Temperature of 850 F

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3. HWO CR THORIUM FUEL CYCLE DEVELOPMENT PROGRAMS

3.1. General

The objective of the HWO CR Thorium Fuel Cycle Development Programs is to demonstrate the performance of thorium fuel elements in an HWO CR environment and to consider the requirements for recycling U-233 and thorium. A preliminary plan for achieving this objective has been developed for each of the candidate fuel elements (metal, oxide, and carbide). These preliminary plans help to guide the selection of a reference fuel element since they provide bases for evaluating the technical feasibility of the designs and the development costs in relation to expected power cost savings. Upon selection of a reference fuel element, the preliminary plan will be the starting point for preparing the detailed program plan to be submitted to the Commission.

The specific tasks of the HWO CR Thorium Program are Task 10 - Reactor Engineering, Task 20 - Fuel Element, and Task 30 - Recycle Technology. Liaison will be maintained with AI-CE to obtain HWO CR technology related to pressure tubes, on-line refueling, reactor control, organic cooling, and major components and systems.

Task 10 - Reactor Engineering, is the coordinating technical task for the program. It includes methods development, optimization studies, fuel management studies, control and safety analyses, economic analyses, and the technical coordination necessary to ensure the compatibility of the thorium fuel elements with the AI-CE HWO CR design. The principal near-term task will be physics methods development, including the analysis of thorium physics experiments performed at BNL and SRL. The need for additional experimental physics data is discussed in section 3.5. The magnitude of work performed under Task 10 is not expected to be affected by the fuel element selected as the reference design.

Task 20 - Fuel Element, is the principal task of the HWOCR Thorium Program. It has the following subtasks:

21. Preliminary Design.
22. Fabrication Development.
23. Fuel Characteristics.
24. Engineering Tests.
25. Fuel Element Assembly Irradiation Test.
26. 1000 MWe Fuel Element Design.

Subtask 21 is part of a continuing effort to perfect and update the reference design by incorporation of the latest findings of the R&D program. Concurrently, irradiation test fuel elements will be designed to demonstrate the features and performance of the reference design. Subtask 26 covers the final design of a thorium fuel element for a 1000 MWe HWOCR and will incorporate the results of all related R&D. The amount of effort required under these design subtasks is relatively independent of the fuel element. The development tasks (22, 23, 24, and 25), however, will significantly depend upon the fuel element chosen as the reference design. These tasks of the fuel element test program are described in relation to each type of fuel element in sections 3.2, 3.3, and 3.4; the costs and schedules for these test programs are compared in section 4.4.

The objective of Task 30 - Recycle Technology, is to develop the reprocessing, feed materials preparation, and fabrication processes for the long-range condition of a mature industry and for the near-term condition of first plants on the line. The related research and development is discussed in section 3.5.

In summary, the research and development required to resolve key technical problems for each type of fuel element is discussed in section 3.6. This information provides the basis for a comparative evaluation of the technical feasibility of the fuel element designs, as presented in section 4.

3.2. Thorium Metal Fuel Element Test Program

3.2.1. Subtask 22 - Fuel Element Fabrication Development

3.2.1.1. Objective

The objective of this task is to develop the fabrication processes for the 1000 MWe HWOCR thorium fuel element,

including the necessary manufacturing procedures and inspection techniques.

3. 2. 1. 2. Course of Action

Present technology will be extended to enable the manufacture of reliable large- and small-diameter fuel tubes for thorium nested-cylinder fuel elements. This task will encompass study and demonstration of extrusion methods, bonding of cladding to fuel, joining of end caps to cylinders, assembly of components, and nondestructive testing. The fabrication development work will provide specimens and fuel elements for fuel characterization and irradiation tests.

3. 2. 1. 3. Status of Technology

Development of nested-cylinder fuel elements fabricated by the coextrusion of Zircaloy-clad thorium-uranium alloys at Hanford Laboratories (HAPO, now Pacific Northwest Laboratories, PNL) and Savannah River Laboratories (SRL) has resulted in the production of satisfactory elements. Nuclear Metals (NMI) cooperated in the SRL work.

Kemper et al. of HAPO¹ described the fabrication procedures used to produce tubular fuel elements of thorium-2.5 wt % uranium-1.0 wt % zirconium clad with Zircaloy-2. These procedures included vacuum consumable arc melting, coextrusion, material evaluation, end closure joining, and cleaning.

The vacuum consumable arc process was used to prepare ingots by double melting. The problem of homogeneity was mitigated by the double melting, the low uranium content, and possibly the zirconium addition. The resulting ingots were 5.75 inches in diameter and weighed up to 200 lb. They were machined into extrusion billets by turning the surface and drilling an axial center hole. Zircaloy cladding and copper canning were solvent degreased, positioned in and around the billet, and electron beam welded.

Coextrusion was performed on a 2750-ton horizontal hydraulic press equipped with a heated extrusion container. The copper canned billet assemblies were appropriately lubricated with colloidal graphite and heated to 760 C (1400 F) for 4 hours. The billets were forward extruded from a 6.09-inch-diameter extrusion container

heated to 427 C (800 F). The opening of the 90-degree cone die was 1.790 inches in diameter, and the mandrel was 1.050 inches in diameter, resulting in an extrusion ratio of 17:1. The extruded tubes, which were 18.5 and 22 feet long, exhibited good surface quality and no evidence of galling.

Dimensional control was found to be entirely satisfactory for HAPO reactor irradiation. Cladding thickness was uniform locally to approximately ± 0.005 inch and averaged ± 0.003 inch for the entire cylinder length. Local fuel thickness tolerance and fissile fuel uniformity were within $\pm 5\%$, and straightness was within 0.010 inch per foot.²

Metallographic examination showed metallurgical bonding at the cladding-core interface; the fuel was partially recrystallized. Subsequent vacuum heat treatment of specimens for 1 hour at 750 C (1380 F), followed by furnace cooling, resulted in uniform diffusion at the interface and a total bond width of 0.0007 inch; the major portion of the bond on the Zircaloy-2 side had the 85.6 atomic % zirconium-thorium eutectoid composition, and the remainder changed linearly to the composition of the fuel alloy. The thorium alloy was almost completely recrystallized, greatly reducing the as-extruded fibrous structure. Random alpha-uranium precipitate was present as expected.

The end closure was made by vacuum brazing with 5 wt % beryllium-zirconium eutectic alloy, which melts at 980 C (1795 F) and brazes at 1050 C (1920 F). Bonding to the thorium alloy was impaired by oxide formation, but was resolved by power brushing the thorium alloy surface with Zircaloy bristles before brazing.

The final step in the manufacture of fuel elements consisted of a 72-hour exposure in deionized steam at 400 C (750 F) and 1500 psig. The assemblies were inspected radiographically, visually, and dimensionally, and the density was determined. It was concluded that coextrusion of thorium-2.5 wt % uranium-1 wt % zirconium clad with Zircaloy on the outside and inside tube surfaces is a practical operation. Reproducible, sound, nonporous, fully bonded braze joints are obtainable on the end closures, and the assembly is suitable for irradiation.

Larson and Huber of NMI and Gray et al. of SRL² described the fabrication procedure for thorium-1.4 wt % uranium

tubes clad with Zircaloy-2. The tubes were coextruded with a 14:1 ratio and had the following dimensions:

OD	2.540 inches
ID	1.830 inches
Cladding	0.030 inch
Length	9 feet

The alloy was vacuum induction melted, and 200 lb ingots were bottom poured into bare graphite molds designed for rapid bottom-to-top solidification to reduce uranium inhomogeneity. The castings were machined into extrusion billets 6.59 inches in OD, 2.69 inches in ID, and 8 inches in length. The carbon content was high, approximately 1350 ppm; the carbon and uranium homogeneity was considered to be good. The fuel billet was enclosed internally and externally in Zircaloy-2 sleeves and end seals; the entire assembly then was copper canned and lubricated with colloidal graphite. The billet assemblies were preheated to 775 C (1425 F) for as much as 20 hours; 9 hours was stated as the preferred soaking time. Extrusion was accomplished in a 2750-ton press; 2000 tons of force was the maximum needed. From their data, the measured dimensions of the coextruded product were found to be typically as follows:

Wall thickness	0.353 ± 0.005 inch
Outer cladding	0.033 ± 0.003 inch
Inner cladding	0.033 ± 0.003 inch
Core	0.286 ± 0.005 inch

Destructive rupture tests were employed, and it was concluded that the Zircaloy-2 cladding was adequately bonded to the fuel for irradiation. Ultrasonic inspection revealed no unbonded regions larger than 1/8 inch, which was the lower limit of detectability for their instrumentation. Radiography was also used to confirm the quality of the coextrusion.

3.2.1.4. Problem Areas

The coextrusion conditions for small- and large-diameter fuel tubing have not been established. It will also be necessary to obtain the characteristics and dimensional tolerances of the product for cylinders of varying dimensions.

To provide adequate control of coolant channel gaps, spacers between fuel cylinders are required. Spacer designs

generated in Task 21 (Fuel Element Preliminary Design) require proof of fabrication feasibility.

Although some development work on methods of applying end closures to coextruded fuel cylinders has been performed, more work is required to gain confidence that the end-closure technique is readily adaptable to production methods and that the end closures will perform satisfactorily at HWOCR operating conditions.

The reliability of present nondestructive testing techniques must be demonstrated to ensure the integrity of the tubular fuel element. Testing techniques are required to determine such items as the quality of bonding between cladding and fuel, the integrity of end-closure joints, and the homogeneity of fuel alloy. It may be necessary to develop new and improved testing techniques to determine these for cylinders having the B&W design.

3.2.1.5. Technical Approach

The development of the fabrication process for nested-cylinder fuel elements will include work on coextrusion development, end-closure development, spacer-rib development, and development of nondestructive test techniques. Detailed task descriptions must necessarily depend to some extent upon the experience and capabilities of the developer. The general requirements are as follows:

1. The coextrusion of thin-walled large-diameter fuel elements (up to approximately 4 inches in diameter and 100 mils in fuel thickness) for a 1000 MWe HWOCR is required. The dimensional tolerances and other characteristics must be suitable for reactor operation. The anticipated requirements of HWOCR nested-cylinder fuel elements are compared with the characteristics of cylindrical fuel elements that have been produced, as follows:

	<u>Thorium HWOCR</u>	<u>SRL/NMI</u>	<u>PNL</u>
Tube OD, in.	4.780 (thorium-optimized-large) 1.218 (AI-CE-small)	2.540	1.790
Tube ID, in.	4.480 (thorium-optimized-large) 0.550 (AI-CE-small)	2.830	1.050
Cladding thickness tolerance (averaged over length), in.	±0.003	±0.003 (outer) -0.001 to +0.005 (inner)	±0.005
Fuel thickness tolerance (averaged over length), %	±3	±3	±5

Thus, HWO CR tolerances generally agree with previous production, but the sizes of the cylinders are not within the range of previous production.

Suggested supplementary areas of evaluation to be performed under this task, after feasibility of producing the cylinders has been demonstrated, might include an evaluation of the two presently available melting techniques and their effect on subsequent fabrication and cost parameters. Both consumable electrode arc melting and vacuum induction melting would be investigated. The casting of large ingots of the fuel alloy and subsequent processing to produce an extrusion billet will be investigated to determine the feasibility of casting a hole in the middle of the ingot. Steps to reduce the amount of machining necessary on the outside and inside surfaces of the ingot will also be investigated since sizeable reductions in scrap production could be realized. Means of producing and controlling the oxide film between the Zircaloy and copper sheaths to prevent interaction during extrusion should also be investigated, as should the effect of carbon content in the fuel alloy upon extrusion characteristics. Integral end closures (see item 4 below) should also be investigated and evaluated.

2. The coextrusion of thin-walled small-diameter (approximately 1 inch) fuel cylinders will be considered since the minimum sized cylinder for the AI-CE design (4.03-inch-ID pressure tube) is in this category, as noted above.

3. An important factor in the operational stability of fuel elements is the fuel-to-cladding bond. The maximum size of unbonded areas, thickness of the bond layer, and mechanical properties of the bond should be established. If necessary, this bond can be improved by heat treatment following extrusion. Such parameters as temperature and soaking time should be studied, therefore, to achieve the best practical bond as determined destructively by metallographic examination and mechanical property testing within the framework of the established requirements. As a further consideration, the dimensional tolerances should not be worsened by heat treatment.

4. The tube ends must necessarily be sealed and spaced relative to the abutting elements. This is accomplished with end caps which must be sealed to the cladding and preferably must be intimately

bonded to the fuel. Development work will be required to determine the best end closure design. There are at least three possible designs: welded, brazed, and integral end caps. The welded or brazed end caps are inserted into the ends of a cut and carefully prepared extrusion. There are problems with surface preparation and braze joint integrity. The integral end caps are formed during extrusion and therefore are well bonded to fuel and cladding. There are problems with the fuel "tailing off" near the ends of the cylinder, which produces variations in fuel content.

5. Nondestructive testing is required to ensure that only reactor quality elements are released for operation. The degrees of bonding, fuel homogeneity, and freedom from defects may be determined by such methods as ultrasonic testing, radiography, autoradiography, gamma absorption, and surface replication. These inspection methods must be evaluated to determine their reliability and sensitivity, and where existing techniques are not suitable, they must be further developed or novel methods must be found to accomplish the required inspection. In those instances where standards are required, they must be fabricated and calibrated.

6. Concentric nesting of the tubular fuel elements of the reference design requires spacers between the elements to maintain the channel width. The spacers must be joined to the cladding, and the design of the spacers must enable the joint to be made and to function as required. The joint must cause only minimal problems in crevice corrosion, fouled organic buildup, fretting corrosion, and mechanical instabilities.

7. The components of the nested fuel element assembly must be designed to allow assembly without introducing undue stresses, elastic instability, blocking channels, or producing other abnormalities. The fuel element assembly must then be perfected by satisfying design requirements and minimizing fabrication costs.

8. The fuel elements fabricated in the course of this investigation shall be used as test specimens as required in other tasks.

3.2.1.6. Summary

A program has been outlined to develop the fabrication processes for a 1000 MWe HWOCR thorium metal fuel element. The experience of PNL and SRL/NMI will be considered in the manufacture and testing of the tubular fuel elements. Thin-walled tubing in the range of 1 to 5 inches in diameter will be coextruded with Zircaloy cladding. Methods for bonding of cladding to fuel, achieving proper end seals, joining spacers to the cladding, and assembling the components will be developed. Nondestructive testing techniques will be adapted to the needs of this program, and necessary standards will be fabricated and calibrated. The fuel elements manufactured in the course of this program will be used as test specimens in other tasks.

3.2.2. Subtask 23 - Thorium Alloy Fuel Characteristics

3.2.2.1. Objective

The objective of this task is to characterize the thorium-uranium alloy fuel by inreactor and outreactor testing.

3.2.2.2. Course of Action

Necessary fuel properties that are not adequately known will be determined or verified under conditions applicable to the HWOCR. The effect of hydrogen upon the mechanical properties of Zircaloy-4, the reference cladding material, is also of significant interest, but will not be treated in depth under this program. Applicable data from related AI-CE, AECL, and B&W company-sponsored³ programs will be evaluated.

3.2.2.3. Status of Technology

Introduction

Thorium-uranium alloys have been studied at a number of laboratories, and a significant body of data exists on their preparation, properties, and behavior under neutron irradiation. Collection and comparison of the information revealed that the investigations were performed on many alloy compositions with differing (or unreported) impurity contents and that the test conditions were not uniform. However, there is sufficient directly applicable information to obviate the need for an extensive investigation.

Phase Equilibria

The thorium-uranium equilibrium phase diagram in the thorium-rich region of interest to this program is disputed to some extent. Bentle of AI⁴ found that the solubility of uranium in thorium was less than 1 wt % at room temperature and increased to about 7 wt % at 1270 C (2320 F). Bannister and Thomson of UKAEA⁵ found the maximum uranium solubility to be 2.7 atomic % at 1343 C (2450 F). Since the reference fuel contains approximately 2% uranium and will not exceed 510 C (950 F), the dispute is essentially academic as far as this program is concerned. At that temperature, the microstructure consists of an alpha-thorium (face-centered-cubic) matrix with a small amount of precipitated alpha-uranium. The first allotropic change is at approximately 650 C (1200 F), where the orthorhombic alpha-uranium transforms to tetragonal beta-uranium. The effect of zirconium on the thorium-uranium system has been investigated by Badaeva and Alekseenko of the USSR Academy of Sciences.⁶ Their ternary diagram shows the lowest temperature "ternary" reaction to be a Class III peritectoid at 605 C (1120 F); modest zirconium additions do not appreciably alter the thorium corner. Since these transformations are well beyond the maximum intended fuel temperature and since the system is probably characterized well enough to enable appreciation of microstructural changes likely to be observed, the performance of additional phase equilibrium studies as a part of this program is not anticipated.

Corrosion

In the event of a defect in the cladding, compatibility of fuel and coolant and more significantly compatibility of fuel and moderator become extremely important. The static corrosion tests of thorium and uranium in orthoterphenyl from 700 to 900 F reported by Pearlman of AI⁷ indicated that "no operating problem would result if there were a cladding failure, with almost any metal alloy fuel." It was stated that the observed corrosion was due to air and/or water dissolved in the organic coolant; however, the amounts present were not given. Additional effort is required in this area.

Although the compatibility of thorium metal with organic coolant is high, the inability of thorium alloys coextruded

with Zircaloy-2 to survive catastrophic failure due to exposure to high-temperature water through a small defect in the cladding was reported by Kneppel of NMI.⁸ A more recent corrosion investigation was performed by Wang⁹ of ANL. He found that the water corrosion resistance of all thorium-uranium alloys from 5 to 40 wt % uranium was poor. The alloying addition of zirconium was beneficial at lower temperatures.

Mechanical Properties

Roll of NMI,¹⁰ Bentle of AI,¹¹ and Farkas of BMI¹² reported the effect of uranium content upon the room temperature mechanical properties of thorium-uranium alloys. The strength increases and the ductility decreases as the uranium content is increased. Bentle^{11, 13} also investigated higher temperature tensile properties. The strength of the alloy at 500 C (930 F) is approximately 50% of the room temperature value. Cole of ATL¹⁴ published the results of an extensive investigation wherein the strengthening effects of zirconium, niobium, molybdenum, carbon, aluminum, and beryllium on thorium-5 wt % uranium and thorium-10 wt % uranium alloy were explored. The effect of heat treatment, rationalized by microstructural studies, was also investigated, and the improvement of mechanical strength due to fine intragranular distribution of alpha-uranium was noted. Leggett of HAPO¹⁵ started the study of the effects of irradiation on the mechanical properties of thorium-uranium alloys with uranium contents to 5.4 wt %. His principal effort with pure thorium¹⁶ expanded upon the earlier efforts of Bement of HAPO.¹⁷ Bement irradiated tensile specimens of two lots of thorium of differing carbon content to 0.1 atomic % burnup (≈ 1000 MWD/MTM). The thorium having higher carbon content continued to possess higher strength and lower ductility after irradiation, since the strength properties increased significantly and the yield point decreased and eventually disappeared with increased exposure. Effective recovery from irradiation damage was obtained by annealing at 750 C (1380 F). Leggett described the mechanical properties of irradiated thorium tested in tension at 590 and 815 C (1095 and 1500 F). Irradiation to 0.06 atomic % burnup (≈ 600 MWD/MTM) doubled the yield strength and reduced the ductility; extensive reduction in area occurred at 590 C (1075 F), whereas little occurred at 815 C (1500 F).

Creep of 5, 10, and 20 wt % uranium-thorium alloys was investigated by Farkas of BMI.¹² He found that the creep resistance of thorium-uranium alloys decreases as the uranium content increases. This was ascribed to the concentration of alpha-uranium at grain boundaries and to the increasing intergranular flow at elevated temperatures. Cole of ATL¹⁴ also investigated creep of 5 and 10 wt % uranium-thorium. As noted above, the strengthening effect and dispersion hardness were explored; zirconium was the most effective additive in improving elevated-temperature mechanical properties.

Irradiation Behavior

Three irradiation experiments were conducted by Kittel of ANL.¹⁸ In the first experiment, wrought thorium and thorium-5 wt % uranium irradiated to 1.5 atomic % burnup ($\approx 15,000$ MWD/MTM) showed isotropic growth and decreased in density at a rate of 1% per 1 atomic % burnup. In the second experiment, wrought thorium and thorium-0.1 wt % uranium, thorium-1.4 wt % uranium, and thorium-5.5 wt % uranium alloys irradiated to 0.3 to 3.6 atomic % burnup (≈ 3000 to $36,000$ MWD/MTM) at irradiation temperatures in the range of 45 to 200 C (110 to 390 F) showed isotropic growth while the densities decreased at rates near 1% per 1 atomic % burnup. In the third experiment, chill-cast thorium alloys containing from 10 to 31 wt % uranium, irradiated from 0.16 to 10.1 atomic % burnup (≈ 1600 to $101,000$ MWD/MTM) at irradiation temperatures from 260 to 1000 C (500 to 1830 F), showed no surface roughening when the uranium was below 25 wt %. The rates of volume increase per 1 atomic % burnup were from 1% at lower temperatures to 2.5% at 650 C (1200 F) and 6% at 800 C (1470 F). These volume changes are significantly less than those reported for the most stable uranium base alloys. Kittel suggested that the excellent resistance to high-temperature swelling of the chill-cast thorium-uranium alloys was due to the very fine uranium particle dispersion, which allowed the fission recoils to escape from anisotropic alpha-uranium into the isotropic and less densely packed thorium matrix. Accelerated swelling was shown to start at about 550 C (1020 F). Ballif¹⁹ of AI irradiated thorium-5.4 wt % uranium alloy and reported values substantially agreeing with Kittel's values.

Neutron irradiation of tubular thorium-uranium alloy fuel elements was performed by PNL and SRL. The specimens and their fabrication are described in section 3.2.1.3.

Irradiation of the PNL specimens commenced in April 1963 in the high-temperature, high-pressure, water-cooled ETR-P7 facility.²⁰ By March 1966 approximately 1.3 atomic % burnup (13,400 MWD/T) had been achieved; the goal is 3 atomic % burnup (26,000 MWD/T). The irradiation temperature varied from 460 to 585 C (860 to 1085 F), and the specific power varied from 46 to 69 watts/gram. The fuel volume increased 2.4% since the start, the first 1.7% being in a generally linear manner and the remainder showing scatter. This gain is slightly less than the theoretical minimum volume expansion due to fission products.

The SRL irradiations were performed in the HWCTR from August to December 1964.²¹ A burnup of approximately 0.35 atomic % ($\approx$ 3500 MWD/MTM) was achieved at a maximum metal temperature of approximately 465 C (870 F). Comparison of dimensions before and after irradiation indicates that the volume change in the region of maximum exposure was about 0.8%.

There is an extreme lack of gas release information. Hayward²² of AI reported gas release values for thorium-11 wt % uranium²³ irradiated to 1.2 atomic % burnup ($\approx$ 12,000 MWD/MTM). Fission gas release averaged about 0.5% for round specimens after burnups of 1.2 to 1.3 atomic % (12,000 to 13,000 MWD/MTM) at about 595 C (1100 F). This release was ascribed to diffusion and recoil; cracked specimens exhibited higher gas release.

Physical Properties

The influence of uranium addition upon the room temperature density of thorium was determined by Blumenthal of ANL.²⁴ The density increased linearly and had a slope change at 1.4 wt % uranium, the limit of solid solubility at room temperature. The 5.4 wt % uranium composition was irradiated by Ballif of AI,¹⁹ and the density was found to be unaffected up to 0.13 atomic % burnup ($\approx$ 1300 MWD/MTM). Farkas of BMI²⁵ also determined densities for varying compositions and reported small differences depending upon impurity content. All three investigations are in substantial agreement.

Lamale of BMI²⁶ determined the effect of neutron irradiation upon thermal expansion of thorium-11 wt % uranium and found a 10 to 35% increase depending upon the test temperature (not the irradiation temperature). Wilson of BMI²⁷ determined the mean thermal expansion coefficient of thorium-2 wt % uranium by X-ray diffraction to 950, 1150, and 1325 C (1742, 2102, and 2417 F).

The thermal conductivity for eight compositions from pure thorium to 20 wt % uranium was calculated by Bentle of AI¹³ using electrical resistivity and the Wiedemann-Franz ratio. The thermal conductivities increased with temperature and decreased with increasing uranium content. The knowledge of thermal conductivity and expansion is considered insufficient for necessary fuel characterization.

Plutonium

The use of plutonium in place of uranium as the fissionable material will probably not introduce new problems. Although the thorium-plutonium-uranium ternary has not been completely defined, work by Blumenthal of ANL²⁸ indicates that, for the temperatures and composition under consideration for the HWOCR, the plutonium will be in solid solution rather than in new or liquid phases. Until uranium is bred into the alloy, the microstructure is monophasic alpha-thorium; upon achieving approximately 1.5 atomic % uranium, gamma-uranium starts to precipitate. If the precipitate is fine and well dispersed, the behavior should be that of the comparable thorium-uranium alloy; however, if the uranium is present in larger aggregates, nonuniform swelling and gas bubbles may result.

3.2.2.4. Problem Areas

The irradiation experiments to date show thorium alloys to be highly resistant to swelling and other deleterious effects of irradiation, especially below 600 C (1100 F); however, data must be obtained for the specific composition and thermal conditions of the HWOCR reference thorium fuel. This added information is necessary to ensure adequate knowledge for optimization of core design. Similarly, thermal conductivity, thermal expansion, and compatibility with an organic coolant of varying moisture content must be definitely established.

3.2.2.5. Technical Approach

The experimental program is divided into two parts, considered separately.

Fuel Alloy Irradiation Behavior

The specific swelling of thorium-uranium alloys as a function of temperature will be investigated. Experimental parameters will include chemical composition, impurity content, and burnup. Attention will also be given to the dispersion of the alpha-uranium phase. Six chemical compositions will be considered:

<u>Uranium, wt %</u>	<u>Zirconium, wt %</u>	<u>Thorium</u>
1.5	--	Balance
2	--	Balance
2.5	--	Balance
1.5	1.0	Balance
2	1.0	Balance
2.5	1.0	Balance

According to Bentle's⁴ equilibrium phase diagram for 1.5 wt % uranium, about 0.5 wt % alpha-uranium should be present at equilibrium below 650 C (1100 F); at 2.0 wt % uranium about 1 wt % alpha-uranium should be present; at 2.0 wt % uranium about 1.5 wt % alpha-uranium should be present. This selection of compositions, therefore, not only brackets the composition of interest, but also allows determination of the effects of varying quantities of alpha-uranium, which is precipitated in the alpha-thorium matrix. The carbon, nitrogen, and oxygen contents will be maintained at values representative of the typical level of contamination. Three temperatures in the range from 400 to 650 C (750 to 1200 F) and three burnup levels, 0.5 to 0.6, 1 to 1.2, and 2 to 2.4 atomic % (5000 to 6000, 10,000 to 12,000, and 20,000 to 24,000 MWD/MTM), will be achieved. It is estimated that the burnup ranges will require approximately 1, 2, and 4 calendar years of irradiation, respectively, in a typical thermal neutron flux provided by a materials test reactor. These specimens will be prepared from stock that has been extruded. The experiment should be performed three times to definitely establish the experimental results and could probably be accommodated in six irradiation

capsules. The effect of irradiation on density and dimensional stability will be determined in this series of tests.

The second phase of the inreactor testing is to determine swelling of the fuel alloy restrained by cladding. The specific fuel swelling will be determined radially and axially on extruded cylinder specimens at various temperatures for various experimental parameters. The chemical composition will be maintained at thorium-2 wt % uranium-1 wt % zirconium. The impurities will be maintained at a constant, typical level. Temperature and burnup will range within the same limits used in the first phase, but only two temperatures and two burnups will be selected. The specimens will be coextruded cylinders approximately 1 to 1-1/2 inches in diameter, 1/8 inch in fuel thickness, and 10 inches in length. Cladding thickness will be 0 (bare), 15, 20, and 30 mils. This experiment will probably be accomplished in eight capsules.

Physical and Corrosion Properties - Outreactor

The mechanical properties are believed to be sufficiently well known for reactor design and need not be further explored at this time, except in conjunction with the normal preirradiation and postirradiation testing of irradiation specimens.

The thermal conductivity of the reference alloy has not been positively established by direct methods. The thermal conductivity will be determined as a function of temperature through the operating range, and the effect of compositional variation will also be investigated.

The thermal expansion must be determined for the operating temperature range and for the expected compositional variation.

The compatibility of the fuel alloy and cladding with the organic coolant has not been fully investigated. A survey experiment is therefore planned to establish this characteristic, which is important for hazards analysis. Specimens of the reference fuel alloy and cladding will be exposed outreactor to flowing organic coolant at temperatures of 150, 260, 370, and 480 C (300, 500, 700, and 900 F) for times ranging from a few hours to several weeks, depending upon

the observed reaction. The experiment will be performed for organic coolant containing low, nominal, and high moisture content. Analysis will be accomplished by metallographic examination and weight change measurements to determine the extent and nature of the reaction.

3.2.2.6. Summary

An experimental program to characterize the reference fuel has been outlined. The following investigations are proposed:

1. The specific swelling under neutron irradiation as a function of temperature will be determined. Experimental parameters will include chemical composition, impurity content, and burnup. The dispersion of alpha-uranium will be considered.
2. A related investigation is the determination of fuel swelling restrained by cladding. Extruded cylinders with varying thicknesses of cladding will be irradiated. The effects of temperature and burnup will be examined.
3. The thermal conductivity will be determined for temperatures through the operating range. The effect of compositional variation will be considered.
4. The thermal expansion will be determined for the operating temperature range.
5. The compatibility of fuel alloy and cladding with the organic coolant will be surveyed at various temperatures and moisture contents.

3.2.3. Subtask 24 - Fuel Element Engineering Tests

3.2.3.1. Objective

The objective of this subtask is to perform outpile engineering tests on the thorium metal nested-cylinder fuel element to obtain information for design optimization and to demonstrate the structural adequacy of the fuel element for operation in an HWO CR. The objective of each test is described below:

1. Thermal Cycling Test is to determine the ability of the fuel element cladding to withstand thermal strain caused by changes in fuel temperature.

2. Hydraulic and Vibration Test is to determine the hydraulic and vibration characteristics of a nested-cylinder fuel element.

3. Structural Loading Test is to determine the adequacy of the fuel element to resist structural loads at reactor operating temperatures.

4. Full-Scale Flow Test is to demonstrate the adequacy of the 1000 MWe HWOCR thorium metal nested-cylinder fuel element to withstand hydraulic and vibration effects.

3.2.3.2. Course of Action

Four basic engineering tests will be conducted:

1. Thermal Cycling Test. The Zircaloy cladding of the cylindrical thorium fuel element will experience thermal strain because of temperature gradients through the cylinder wall and differential expansion between the fuel and the cladding. Specimens will be thermally cycled until a predetermined number of cycles or fatigue failure occurs.

2. Hydraulic and Vibration Test. Flow testing of fuel elements (representing an interim design) will be performed in low-temperature and low-pressure water. Flow testing is required for fuel elements to be irradiated in WR-1. Information gained from these tests will also be used in the fuel element design (Subtask 21).

3. Structural Loading Test. A test will be performed at elevated temperatures to determine the effect of long-time column loads on the fuel cylinders.

4. Full-Scale Flow Test. A flow test of the 1000 MWe HWOCR thorium fuel element will be performed outpile in a hot organic loop.

3.2.3.3. Status of Technology

Limited testing of thorium nested-cylinder fuel elements has been done at PNL and SRL. This test work has been aimed primarily at determining irradiation behavior. Tests similar to those proposed for the HWOCR have not been performed on thorium cylinders. The specific status of the tests is as follows:

1. Thermal Cycling Test. For some time PNL has attempted to derive equations to analyze the effects of thermal cycling on the integrity of the junction between the end cap, the cladding, and the fuel. So far complete success has not been attained. To our knowledge, no testing has been performed in this area.

2. Hydraulic and Vibration Test and Full-Scale Flow Test. Some limited flow testing of thorium-uranium fuel assemblies (10 feet long) has been performed at SRL (DP 943). The assemblies consisted of only one fuel cylinder and inner and outer housing tubes (not bearing fuel). Flow testing was accomplished in 260 C water having a flow rate of 150 gpm. This flow rate is considerably lower than that expected in the thorium-fueled HWOCR, and results obtained from the SRL test may not be applicable to organic-cooled HWOCR fuel elements.

It is well known^{29, 30, 31} that high-velocity coolant flow parallel to a fuel element assembly may induce vibrations in the assembly. This phenomenon is not well enough understood to satisfactorily determine vibratory response by using only analytical techniques; therefore, testing is considered essential. Unpublished B&W-sponsored R&D in this area further substantiates the need to investigate the phenomena of parallel-flow-induced vibration.

3. Structural Loading Test. Structural proof tests of the type proposed have not been performed on components of coextruded nested-cylinder fuel elements.

3.2.3.4. Problem Areas

Various mechanical, thermal, and hydraulic problems in the nested-cylinder design are better solved by experimentation than by analytical techniques. These problems are outlined below:

1. Thermal Cycling Test. Temperature gradients and thermal differential expansion in the end cap region of the cylindrical fuel element will cause thermal strains in the Zircaloy cladding. Because of power transients and reactor shutdowns, strain cycling will take place in the fuel element cladding. If the amplitude of the cycling is too severe or the number of cycles is too great, cladding failure may take place. Thermal strain limits must be known to design against failure.

2. Hydraulic and Vibration Test and Full-Scale Flow Test.

Hydraulic parameters of the thorium nested-cylinder fuel element are not established, and such items as flow distribution and pressure drops are required. Vibration of the fuel element components (individual cylinders) is not considered to be a problem if there is no backlash between nested cylinders. However, vibration of the fuel element assembly and the associated fretting and wear against the pressure tube may be a problem.

3. Structural Loading Test. Axial loads are imposed on the fuel elements by the weight of the fuel column, by hydraulic loads, and by a preload applied by the structural supports for the fuel assemblies within the pressure tube. Analysis has shown that it is desirable to carry the axial loads on the outer cylinders since they have the best structural integrity as a column. The effects of these long-time sustained loads on the creep buckling behavior of the fuel element need investigation.

3.2.3.5. Technical Approach

Representative test fuel elements will be fabricated for use in the engineering tests. Suitable test facilities can be provided at B&W's Alliance Research Center. Information gained from the tests will be used to ensure the adequacy of the fuel elements and to further refine the fuel element design.

1. Thermal Cycling Test. Coextruded cylindrical specimens, including end caps, will be thermally cycled (outpile) from 300 to 850 F until signs of cladding fatigue are detected or until the maximum feasible number of cycles has been accumulated. Time-dependent and irradiation effects will probably have a significant influence on the ability of the cladding to withstand thermal cycling. Therefore, some thermal cycle testing may be performed on cylinders fabricated to simulate as closely as possible the expected material effects of irradiation and time. Approximately six specimens will be tested. Different specimen sizes will be included in the total number of specimens to be tested; the actual sizes will be determined later.

2. Hydraulic and Vibration Test. Fuel element assemblies will be tested in a water loop to determine the hydraulic and vibration characteristics of the interim fuel element design prior to irradiation testing. A water loop will be used to simplify the testing and reduce the cost. Adequate extrapolation of water environment test results can be made to apply to the HWOCR organic coolant operating conditions.

Flow velocities in the range of 30 to 45 fps and temperatures of about 180 to 200 F will be used. Later flow tests of the 1000 MWe HWOCR fuel element will be performed in organic coolant at reactor operating conditions. The axial load expected on the fuel element will be simulated in the test. The determination of the resonant frequency of individual fuel cylinders by bench testing or shaker testing is not presently contemplated for this type of fuel element. The test will determine the flow distribution between the various annuli, the entrance and exit losses for the fuel element, the pressure drop across the fuel element, the effect of the spacers, the degree of vibratory response of the assembly, and some fretting and wear characteristics of the fuel element and pressure tube combination. Two fuel elements will be required for testing.

3. Structural Loading Test. The need for this test will be based on the results of analytical investigations of the fuel element creep buckling characteristics and creep testing of the fuel material proposed for Subtask 23. If testing is necessary, one cylinder of the fuel assembly will be subjected to loads representative of those expected during reactor operation. The loaded cylinder will be heated to a temperature between 800 and 1000 F and maintained at this temperature for about 12 to 18 months. At the conclusion of this time, measurements will be made to determine the amount of creep buckling that has taken place in the fuel cylinder.

4. Full Scale Flow Test. Fuel element assemblies will be tested in a hot organic loop to determine the hydraulic and vibration characteristics of the 1000 MWe HWOCR fuel element design. An organic loop will be needed to confirm that the fuel element design is adequate for operation at simulated HWOCR conditions. Flow velocities in the range of 30 to 45 fps, temperatures of about 750 to 850 F, and pressures of about 400 psi will be used.

3. 2. 3. 6. Summary

Engineering tests will be performed to obtain information necessary for design optimization and to demonstrate the adequacy of the fuel element for operation in a 1000 MWe HWO CR. Tests to be performed include the thermal cycling test, hydraulic and vibration test, structural loading test, and full-scale hydraulic flow test.

3. 2. 4. Subtask 25 - Thorium Alloy Fuel Element Assembly Irradiation Tests

3. 2. 4. 1. Objective

The objective of this subtask is to determine the operational behavior of the reference fuel element assembly under conditions closely simulating those of the HWO CR.

3. 2. 4. 2. Course of Action

The fuel element assemblies will be irradiated in a succession of at least five experiments, progressively developing confidence in the behavior of thorium alloy nested-cylinder fuel elements for the HWO CR. The final step will be a proof test in the WR-1 under conditions as closely representative of the HWO CR as are reasonably attainable.

3. 2. 4. 3. Status of Technology

Prior irradiations of thorium-uranium alloy cylinders are discussed in section 3. 2. 2. 3.

3. 2. 4. 4. Problem Areas

The major problems that need to be investigated are (1) the dimensional stability of the fuel element during irradiation, (2) the mechanical and structural integrity of the fuel element during irradiation, and (3) the effect of organic fouling during irradiation on the thermal and hydraulic performance of the fuel element.

3. 2. 4. 5. Technical Approach

A number of multicylinder fuel element assembly irradiation tests will be performed in the WR-1 reactor to obtain information on operational performance at simulated HWO CR conditions.

A preliminary evaluation previously determined the feasibility of irradiating in the WR-1 an existing thorium-1.5 wt % uranium fuel tube available from the HWCTR program at SRL. A more detailed evaluation will determine the practicality and desirability of irradiating this tube, but this experiment is not considered an integral part of the test program discussed below.

The first irradiation experiment in the WR-1 reactor is a demonstration of the feasibility of the multicylinder design. This initial inreactor investigation will necessarily reflect the tubular designs developed at PNL and NMI/SRL. The multicylinder design will have appropriate length and diameter for the WR-1. This fuel element will be designed for operation at reduced power. The operational characteristics will be studied and will provide design information for further testing. The specific information obtained will include power and temperatures achieved and dimensional stability. From destructive analysis, the gas release, corrosive effects (including the extent of hydriding of the cladding), and other physical and microstructural observations will be obtained.

The second irradiation experiment in the WR-1 reactor is similar to the first except that power will be increased and physical and functional design features applicable to the HWOCR will be incorporated. Information from the first experiment will be considered as it is received. Since the programs will overlap in time, the full benefit of experience with the first experiment will not be realized in the second.

The third irradiation experiment in the WR-1 reactor is similar to those preceding except that the improved design will be based on the experiences of the first two experiments. The results of this experiment will help establish the prototypic design parameters for the HWOCR.

The fourth irradiation experiment in the WR-1 reactor will test the operational performance of a near-prototypic fuel element modified in diameter to be accommodated in the WR-1. Defect testing is also planned.

The fifth irradiation experiment in the WR-1 reactor will continue to test the operational characteristics of a reference

design fuel element. Information obtained from the fourth irradiation will be incorporated in this experiment. The results obtained will contribute to the design of the reference 1000 MWe HWOCR thorium fuel element.

If the need for additional irradiation experiments is indicated, they will be further described in amendments to this document. It is presently not feasible to predict needs arising from future experimental work.

3.2.4.6. Summary

The planned irradiation program will determine the operational behavior of the reference fuel element assembly under conditions closely simulating those of the HWOCR. At least five irradiation experiments will be performed in the WR-1 reactor. They will start with initial irradiations of a multicylinder fuel element assembly based on current technology and culminate in the irradiation of a prototype of the reference fuel element assembly. The results obtained will contribute to the final design of the 1000 MWe HWOCR thorium fuel element.

3.3. Thorium Oxide Fuel Element Test Program

3.3.1. Subtask 22 - Thorium Oxide Fuel Element Fabrication Development

3.3.1.1. Objective

The objective of this subtask is to develop the fabrication processes for the 1000 MWe HWOCR thorium oxide fuel element, including the necessary manufacturing procedures and inspection techniques.

3.3.1.2. Course of Action

The present technology for thorium oxide is more advanced than that for thorium metal or monocarbide; therefore, less development effort will be required to produce acceptable oxide fuel elements. The work will be aimed primarily at fabricating fuel pins of the desired shape and size with acceptable and reproducible densities and fuel mass variations per unit length of fuel pin. The

fabrication development will provide specimens and fuel elements for fuel characterization and irradiation tests.

3.3.1.3. Status of Technology

The vibratory compaction process has been investigated over the past few years at many laboratories, such as HAPO (now PNL),³²⁻³⁵ GE at San Jose,^{36,37} ORNL,³⁸⁻⁴⁰ Combustion Engineering,⁴¹ SRL,⁴² and B&W.⁴³⁻⁴⁵ In addition, Euratom⁴⁶ at several European laboratories has sponsored work involving the Vipac process; there have also been recent Japanese⁴⁷ investigations into the Vipac process. Most workers have particular methods and innovations peculiar to their specific needs and equipment; therefore, correlation of results is often difficult. For this reason, the particular parameters of interest (for example, height and weight variations of the fuel column, density and fuel mass variations) must be established for the particular process, feed material, and fuel pin size and shape used.

Basically, fabrication of fuel pins by vibratory compaction consists of the following steps:

1. Feed material preparation including synthesis and sizing of the dense particles.
2. Introduction of fuel powder into the fuel tube with vibration.
3. Further compaction of the fuel column by vibration.
4. Measurement of fuel column height and weight.
5. Final end capping and nondestructive testing of weld integrity.
6. Nondestructive measurement of fuel mass variations per unit length of fuel pin.
7. Cleaning, final dimensional inspection, and identification.
8. Assembly of pins into fuel elements.

Fuel tubes having one end cap welded and inspected are introduced as one of the incoming components for the process.

Based on present procedures and fuel feed materials used at B&W, it is anticipated that densities in the range of $87 \pm 1\%$ of theoretical density could be maintained and that fuel mass

variations (based on a 1-inch scan length as the unit length) of $7 \pm 3\%$ could be achieved. Fuel column height and weight variations commensurate with the density range would be on the order of ± 1 inch and ± 20 grams, respectively. All of these parameter ranges are based on experiments with fuel pins 10 feet long and 0.390 inch in inside diameter.

3.3.1.4. Problem Areas

Work would consist of vibratory compacting an acceptable fuel feed material sized to an appropriate dimension for the particular HWOCR oxide pin size and shape and establishing the density and mass variation to a statistically dependable value. Irradiation specimens shorter than approximately 18 to 24 inches present a particular problem in fabrication since it is difficult to scale down the exact physical parameters that can be obtained in the larger pins; experimentation would therefore be necessary to establish the physical parameters in small capsule irradiation specimens. No major fabrication problems are expected as a result of using SAP cladding.

3.3.1.5. Technical Approach

This subtask, which can be performed at B&W, is aimed at developing statistically reliable values for the various fabrication-related physical properties in the HWOCR fuel pins that affect reactor performance. As previously discussed, these parameters are not anticipated to be greatly different from those established on fuel pins for pressurized water power reactors. The program can be subdivided into three parts: (1) preparation of suitable feed material, (2) optimization of physical parameters (density, mass variation, particle size distribution) in the full-sized HWOCR fuel pin (42 inches long by 0.447 inch in diameter by 0.020 inch of SAP cladding) produced by compaction, and (3) optimization of physical parameters in the small capsule irradiation specimens.

To achieve high fuel column densities, the feed material required for vibratory compaction must have a high particle density (greater than 98% of theoretical) and must not have excessive sorbed gas content. There are four possible techniques to obtain this type of material; arc fusion, crushed slugs, sol-gel, and pneumatic impaction. The latter three are the most economical; the sol-gel

technique has the greatest technical advantage at the present time. It has been demonstrated⁴⁰ that the sol-gel and vibratory compaction techniques can be successfully coupled in a semiremote facility. Recent B&W experience has shown that this process can also be used in a glove-box type of operation. A complete characterization of the fuel material, as produced for the HWOCR program, is necessary however. The sorbed gas content must be well known and low, and the uranium-to-thorium ratios and O/M ratio must be very well known in the small quantity of powder produced for the irradiation experiments.

It will be necessary to obtain, on a statistical basis, routinely achievable characteristics such as bulk fuel density, mass variation, and column height and weight values and variations. These data will be used to realistically set future production variables. A pilot run of from 200 to 400 pins will be required.

When fabricating fuel pins less than 1 to 2 feet long by the Vipac process, difficulty is encountered in obtaining properties equivalent to those obtained in longer pins. The primary reason is that the zone of interaction, where the various powder particle sizes fit together into the proper packing, is generally longer than 8 to 10 inches. It is necessary that this zone be operable to achieve uniformly high densities. In shorter pins this zone is abbreviated to the point of having negligible effect, depending upon the length of the pin. Some work will be necessary to obtain properties for the small capsule irradiation specimens that would be equivalent to the properties for the longer, full-sized HWOCR fuel pins.

Fabrication of pins into fuel elements will require a regular mechanical assembly procedure utilizing applicable techniques developed under the AI-CE program. Additional fabrication development is not planned except as required to adapt assembly methods to equipment at B&W.

Procedures developed in this subtask will be made compatible with recycle and remote handling methods (Task 30) and will permit expansion to full-core production quantities.

3.3.1.6. Summary

A program has been outlined for the development of thorium-dioxide-fueled, vibratory-compacted pins. Sol-gel

material is considered to be most applicable. It will be vibratorily compacted in SAP tubes, and the necessary production variables will be established.

3.3.2. Subtask 23 - Thorium Oxide Fuel Characteristics

3.3.2.1. Objective

The objective of this subtask is to characterize thoria-urania fuel by inreactor and outreactor testing.

3.3.2.2. Course of Action

Necessary fuel materials properties that are not adequately known will be determined or verified under conditions applicable to the HWOCR. Investigations of the SAP cladding material will be performed by AI-CE, except when specifically related to thoria-urania fuel.

3.3.2.3. Status of Technology

Thermal Conductivity

The thermal conductivity of thoria-based nuclear fuels has been investigated only to a limited extent, probably not sufficient for completely accurate design or reliable prediction for the reactor lifetime.

For thoria, or thoria with less than 15 wt % urania, the thermal conductivity has a relatively low value, characteristic of refractory oxide materials.^{48,49} The thermal conductivity up to the melting temperature has not been evaluated to date because of limitations in experimenting at high temperatures. Data obtained below about 1400 C (2550 F) have shown that thermal conductivity decreases with increasing temperature; 0.03 watt/cm-C (1.73 Btu/hr-ft-F) is a typical value at 1400 C (2550 F). The conductivity also decreases with increasing urania concentration and increasing bulk body porosity.

Some questions exist regarding the closeness to stoichiometric composition of the thoria-urania specimens utilized in some of the reported thermal conductivity investigations. For example, in Kingery's⁵⁰ work, the specimens were fabricated by air

firing for several hours at 1540 C (2805 F). Kingery indicated the composition of his specimens as $(\text{Th}_{0.9}\text{U}_{0.1})\text{O}_{2+x}$, showing oxygen content beyond the stoichiometric composition due to heating in air above 500 C (930 F). Increased oxygen content lowers the thermal conductivity.^{51,52}

The thermal conductivity of compacted powders, like that of presintered pellets, is the sum of heat transfer by lattice conduction and radiation. However, the lattice conduction contribution is reduced because of the loss of fuel lattice matrix continuity. Heat is transferred by gaseous conduction across the spacings between powder particles and directly at the points of particle contact. The gas is considered the continuous phase. At higher temperatures, the powder sinters to form a porous mass and undergoes microstructural alteration similar to that of presintered pellets at these higher temperatures; therefore, the effective thermal conductivities of the sintered "powder" and pellet are similar. However, the average integrated thermal conductivity of the powder fuel pin would remain lower in value than that of the pellet because of the insulating effect of the unsintered powder in lower temperature regions of the fuel pin.

In a study of the thermal conductivity of powder,⁵³ several theoretical models were considered. Excellent agreement was found between a model suggested by Godbee⁵⁴ and experimental values for magnesia, alumina, and zirconia powders in air at atmospheric pressure and temperatures from 95 to 815 C (200 to 1500 F). However, this model is applicable only to a well characterized powder (having known bulk thermal conductivity and particle size, shape, and distribution) and to a gas system for which the physical constants are established.

The parameter, $\int_{T_1}^{T_2} \lambda d\theta$, known as the heat rating, has been used⁵⁵ to describe the integrated thermal conductivity of UO_2 . (Thermal conductivity values are obtainable by differentiation of the heat rating with respect to temperature.) The integral limits (temperatures) can be obtained by noting the temperature required to cause observed microstructural changes. However, no definitive correlation has been established for thorium-urania, although a start has been made in this direction.⁵⁶ Irradiation studies⁵⁷ of presintered thorium-6.5 wt % urania pellets indicate that the high-temperature thermal conductivity of

thoria-urania is the sum of lattice conductivity and radiant energy transfer, similar to that observed for urania. This result was developed from the observation that the onset of fuel-void redistribution (structural alteration by columnar grain formation) by a sublimation-condensation mechanism was indicated to have occurred at about 2400 C (4350 F). It is interesting to note that Neimark⁵² calculated that the occurrence of this microstructural change in pure thoria was between 2150 and 2400 C (3900 and 4350 F). He used vapor pressure data and the urania vapor pressure-sublimation model.⁵⁸

Rao⁵⁹ of AECL has reported $\int_{400\text{ C}}^{\text{melt}} \lambda d\theta$ values of 67 to 77 watts/cm (26 to 30 kw/ft) for ThO₂-5 to 19 wt % UO₂ pellets having 95% of theoretical density. However, pellets of ThO₂-2.7 wt % UO₂ did not melt or form columnar grains at 72 watts/cm (28 kw/ft). Bain of AECL⁶⁰ obtained an $\int_{400\text{ C}}^{2150\text{ C}} \lambda d\theta$ value of 57.5 watts/cm (22 kw/ft) for ThO₂-2.2 wt % UO₂ pellets having 95% of theoretical density.

Based on these data, an $\int_0^{\text{melt}} \lambda d\theta$ versus temperature plot was constructed by Gross and Harbinson⁶¹ for ThO₂-2 wt % UO₂ pellets. Kingery's⁵⁰ low-temperature thermal conductivity data were taken for the low-temperature lattice conduction component and expressed as $\int_0^{800\text{ C}} \lambda d\theta$. Bain's value of $\int_{400\text{ C}}^{2150\text{ C}} \lambda d\theta$ was appropriately plotted on the $\int \lambda d\theta$ curve and was used to interpolate between 800 to 2150 C; the curve was then extrapolated to the melting temperature. This results in an $\int_0^{\text{melt}} \lambda d\theta$ value of 148 watts/cm.

A further inferred analysis was performed to construct the $\int_0^{\text{melt}} \lambda d\theta$ curve for vibratory compacted ThO₂-UO₂ powders. A 50% reduction was made in the lattice conduction component up to 1900 C (3450 F), selected as the "sintering temperature;" beyond this temperature the pellet expression was used. This procedure was found to be satisfactory for urania⁶² and was therefore adopted for thoria-urania. This results in an $\int_0^{\text{melt}} \lambda d\theta$ value of 97 watts/cm.

Results of irradiation experiments with ThO₂-5 wt % UO₂ vibratory compacted powder conducted by Olsen et al. of ORNL⁶³ tend to support the previous analysis. Test rods were operated at linear heat ratings of 1000 watts/cm (30.5 kw/ft) where $\int_0^{\text{center}} \lambda d\theta$ was 100 watts/cm without central melting. The linear heat rating was sufficient to produce central voids and columnar grain growth. Olsen⁶⁴ also stated

that a 40% higher power rating would be required to produce similar microstructural changes in ThO_2 -5 wt % UO_2 compared with UO_2 . A typical value of $\int_0^{\text{melt}} \lambda_{\text{d}\theta}$ for UO_2 powder is 65 watts/cm;⁶² applying Olsen's 40% correction would increase $\int_0^{\text{melt}} \lambda_{\text{d}\theta}$ to 91 watts/cm for ThO_2 -5 wt % UO_2 . Clearly, additional investigation is required.

Another problem in the understanding of the thermal conductivity of oxide fuels is the effect of porosity. Numerous relationships have been proposed⁶⁵ to correct thermal conductivity of test specimens to that of theoretically dense material. A study is being pursued at Bettis Atomic Power Laboratory⁶⁶ to quantitatively analyze the correction factor. The effects of such variables as pore shape, pore size, and orientation are being evaluated. Publication of this study is anticipated by the end of 1966.

Fission Gas Release

The only fission gas release data available for thorium-uranium powder are from the ORNL sol-gel experiments reported by Olsen.⁶³ Fuel pins containing sol-gel and arc-fused vibratorily compacted ThO_2 -4.5 wt % UO_2 were irradiated to burnups of 12,000 to 81,000 MWD/MTM (1 to 6 atomic %) at peak linear heat ratings of approximately 325 watts/cm (10 kw/ft); $\int_0^{\text{center}} \lambda_{\text{d}\theta}$ was 32 to 38 watts/cm. Based on Kr-85 analysis, slightly higher gas release is indicated for the sol-gel material than for the arc-fused material (17% compared to 7%). According to Olsen, the principal factor in gas release at these heat ratings is the time at temperature, which behavior is expected with a diffusion-controlled mechanism.

To study the effect of high heat ratings, ORNL irradiated vibratorily compacted ThO_2 -5 wt % UO_2 sol-gel powder fuel pins at peak linear heat ratings of 1000 watts/cm (30.5 kw/ft) where $\int_0^{\text{center}} \lambda_{\text{d}\theta}$ was 100 watts/cm to burnups beyond 20,000 MWD/MTM (1.3 atomic %). The gas release values were less than 30%. There were indications that the calcining atmosphere (in the preparation of the oxide powder) influences the release rates. Fuel that had been calcined in air showed a 39% Kr-85 release compared to 28% for material calcined in the standard atmosphere of argon and 4% H_2O . The linear heat ratings were sufficient to produce central voids and columnar grain growth.

Investigations are required to establish positively the fission gas release properties of thoria-urania powder and to determine the pressure buildup during reactor operation.

Chemical Compatibility

The chemical compatibility between ThO_2 -2 wt % UO_2 and SAP may be viewed as two separate problems. First, the alumina dispersed in the SAP and present as surface oxide film must be compatible with the oxide fuel. Second, the aluminum matrix must also be reasonably nonreactive with the fuel at the temperatures of interest. In the current fuel design the fuel surface temperature is approximately 650 C (1200 F) at 100% power, the cladding inner temperature being approximately 470 C (880 F). At these temperatures, solid-state reactions between the oxide phases are extremely unlikely and certainly are not indicated in existing equilibrium phase diagrams.^{67, 68}

The HWOCR SAP Handbook⁶⁹ states that " UO_2 has been shown to be compatible with XAP001 (ML57) at 1100 F for 5000 hours."⁷⁰ Considering this statement and the rather low concentration of UO_2 in the reference oxide fuel, the UO_2 constituent is not expected to present a compatibility problem. Reduction of ThO_2 by aluminum to form alumina and thorium is energetically impossible at these temperatures (in straightforward solid-state reactions), since the free-energy change for the process is over 40 kcal/mol of Al_2O_3 (positive).⁷¹ At SRL⁷² a mixture of aluminum powder and ThO_2 powder (or UO_2 powder in other experiments) was baked for 16 hours at 575 C (1065 F) in a vacuum and was then extruded at 450 C (840 F) in an aluminum can. Although not specifically discussed in the report, macrophotographs shown do not reveal any signs of gross chemical attack or void formation due to melting or diffusion. However, Myrick and Folger of SRL,⁷³ utilizing the results of Raleigh of Al,^{74, 75} found that fine particles of aluminum and thorium were capable of reacting to form ThAl_3 and alumina. In the solid state, the threshold temperature was given as near 560 C (1040 F). For the situation existing in the HWOCR, where contact between the reactants is not nearly as intimate as that between the mixed powders and where the temperature upon contact is lower (the aluminum constitutes a heat sump for the fuel), the probability of the reaction occurring to any large extent is small.

Phase Equilibria

A critical evaluation of ThO₂-UO₂ phase equilibria studies⁷⁶ by Hoekstra of ANL and Parker of NBS shows that the equilibrium phase is a solid solution over the entire composition range and that the lattice constant obeys Vegard's Law. The crystal structure is face-centered cubic.

Plutonium

The use of plutonium dioxide in place of uranium dioxide as the fissionable material will probably introduce no new problems. Irradiation studies conducted at Hanford Laboratories (now PNL)⁷⁷ indicate that hydrogen-sintered (Th, Pu)O₂ solid solution pellets in the composition range of interest for thermal reactor fuels (up to 5 wt % PuO₂) exhibited satisfactory irradiation performance to a burnup of 2.4×10^{20} fissions/cm³ ($\approx 10,000$ MWD/MTM). During irradiation, (Th, Pu)O₂ recrystallized in the manner usually observed with UO₂. Microstructures developed during irradiation were similar to those developed in UO₂ and UO₂-PuO₂. It was deduced from a comparison of these structures that, in the low (<5%) PuO₂ specimens, the power output would be 20 to 30% higher than that at the same fuel temperature in UO₂. Further work will be directed toward determining gas release characteristics and fission product migration and identification. ORNL⁶³ has irradiated three fuel rods containing sol-gel ThO₂ mixed with PuO₂ before calcining. The rods were loaded by tamp packing. After exposure at linear heat ratings as high as 245 watts/cm to burnups of 29,000 MWD/MTM, the fission gas release rates were less than 5% and the microstructures were similar to those of ThO₂-UO₂ exposed under similar conditions.

PuO₂ and UO₂ are isomorphous and otherwise chemically and physically similar.⁷⁸ It would be expected that the substitution of plutonium dioxide for uranium dioxide in the thorium dioxide matrix in the concentrations required would not introduce substantial changes in the overall physical and chemical situation. The quaternary [(Th, Pu, U)O₂] developed due to breeding in of uranium would probably behave as the ternary in the region of interest.

3.3.2.4. Problem Areas

Further investigation is required to determine the thermal conductivity and gas release properties of the reference fuel and to determine the compatibility of fuel and cladding. The thermal conductivity investigation is particularly troublesome due to the possibility of large errors in the determination, the high temperatures that are necessary, and the added difficulty of working with powder. Internal gas pressure buildup is particularly important in the HWOCR because of the SAP cladding and the low external pressure on the fuel pin.

3.3.2.5. Technical Approach

Thermal Conductivity

The determination of the thermal conductivity of ThO_2 -2 wt % UO_2 vibratory compacted powder will require a comprehensive treatment if meaningful results are to be obtained. Inreactor and outreactor determinations, using both pellets and powder, will be necessary to firmly establish this property. Experimental determinations will be performed on two compositions, the reference composition and another composition having a greater uranium content.

Conventional thermal conductivity determinations will be performed on pellet material to 2000 C (3630 F) to establish the plot of thermal conductivity in this temperature range. This "pellet" curve will serve as a reference for the more difficult to establish "powder" curve. Outreactor thermal conductivity determinations will be performed on powder material to the highest attainable temperature. This will establish the plot of thermal conductivity in this temperature range for powder. Cognizance must be taken of sintering of the powder and the effect of irradiation upon conductivity in the application of these data.

Both pellet and powder fuel will be irradiated to central temperatures not exceeding 2000 C (3630 F). These specimens will be instrumented with thermocouples for temperature determination. Other specimens will be irradiated at sufficiently high power ratings to cause center melting, thereby producing a fiducial point. The radial location at which melting is first observed will be considered as operating at the melting point of the material. The radius of melting

will be determined by metallographic observation of structure alteration and/or observation of melting in metal fuses properly located in the fuel body. Using burnup measurements, the heat rating $\left(\int \lambda d\theta\right)$ will be obtained for the temperature of melting. This result is one point on the plot of heat rating versus temperature. The general shape of this plot may be obtained from consideration of uranium dioxide behavior. The outreactor thermal conductivity data expressed as heat rating will provide the lower end of the plot; gaps will be filled by interpolation. Differentiation of the curve of heat rating versus temperature will provide the curve of thermal conductivity versus temperature.

These irradiations will be brief, approximately three days, during which time the power levels of the specimens will be maintained as constant as practicable. This is necessary to minimize the error in calculating heat rating from fuel burnup determinations. It is anticipated that approximately 12 capsules will be required to accomplish the objectives of this experiment. Each capsule will contain approximately three specimens.

Gas Release

Internal pressure in a fuel element is a complex function of the initial fill gas introduced with the fuel, its removal by reaction with the fuel or cladding material, and the release and reaction of sorbed and fission gases during operation. The release of sorbed and fission gases is a function of temperature and the microstructure and macrostructure of the fuel. There are two areas of specific interest: (1) the internal pressure of the fuel versus time (dynamic gas release) and (2) the quantity and species of the released gas at a preselected burnup (static gas release). Both static and dynamic gas release experiments are required to obtain sufficient understanding of the process to confidently design for gas release throughout the core life.

The dynamic gas release experiment will, by inreactor monitoring, determine the internal gas pressures of a series of $\text{ThO}_2\text{-UO}_2$ fuel pins during reactor operation. Pressure determination will require pressure transducers capable of reliable operation in an irradiation environment; a mechanical null-balance diaphragm switch

is proposed. Capsules will be irradiated for approximately 60 days. Gas pressure and neutron flux will be recorded as a function of time.

One set of specimens will be subjected to an initial high power to establish an altered grain structure. Another set of specimens will be subjected to a terminal high power to realter the grain structure after a substantial amount of fission gas has been generated. Gas release values of the two series of specimens will be compared to determine the significance of variations in power level over the operating life of a fuel element on the gas release characteristics of pellet and powder thorium-uranium fuel. Power bursts will exceed normal power to the degree required for extensive recrystallization or center melting. It is anticipated that eight capsules, each containing one specimen, will be required for this experiment.

The static gas release experiment will produce data from postirradiation sampling to develop a relationship between gas release and heat rating $\left(\int \lambda d\theta\right)$. Experimental parameters will be selected so that the operating heat rating varies up to that required for center melting. Both powder and pellets having two different UO_2 compositions will be irradiated. Gas release data obtained from pellets will be used for correlation of the results of this experiment with those from other installations and to establish the general shape and magnitude of the curve representing the relationship between gas release and heat rating for these compositions.

The capsules for this subtask will be irradiated for at least 6 months—the time required for sufficient fission gas buildup to permit analysis with some precision. Capsule instrumentation will consist of neutron monitors and thermocouples where practicable. Approximately 10 capsules will be required to accommodate the specimens having powder and pellet form, two compositions, and perhaps four heat ratings. Postirradiation examination of the specimens will determine the relationship between gas release and heat rating for pellet and powder materials for the two uranium compositions investigated. Analysis of characterization and operating data, gas sampling, and radiochemical, isotopic, and ceramographic examinations will be accomplished.

Compatibility of Fuel With Cladding

The compatibility of thoria-urania powder with SAP will be considered from both mechanical and chemical viewpoints. A number of the specimens in the preceding gas release subtask will be SAP clad, thus providing information regarding chemical compatibility of the SAP with thoria-urania. By using varying wall thicknesses, the ability of SAP cladding to withstand gas pressure and its fuel restraint capability will also be determined.

3.3.2.6. Summary

An experimental program to characterize the reference thoria-urania fuel has been planned. The following investigations are proposed:

1. Thermal conductivity will be determined in a comprehensive treatment utilizing both inreactor and outreactor experiments. Powder and pellet materials will be examined; the "pellet" curve will serve as a reference for the "powder" curve, which is more difficult to establish.
2. Gas release will be studied by two procedures: determination of internal pressure (inreactor) versus time and determination of the quantity and species of the gas released at a preselected burnup.
3. Compatibility between fuel and cladding will be determined. Both chemical and mechanical compatibility in a neutron environment will be considered.

In all instances parameters will be selected to obtain information necessary for confident HWOCR core design.

3.3.3. Subtask 24 - Thorium Oxide Fuel Element Engineering Tests

3.3.3.1. Objective

The objective of this subtask is to perform outpile engineering tests on the thorium oxide clustered-pin fuel element to obtain design and performance data to ensure the structural adequacy of the fuel element for operation in an HWOCR.

3.3.3.2. Status of Technology

The basic engineering tests for the development of a SAP-clad clustered-pin fuel element are being performed under the AI-CE HWOOCR program, including a SAP development program at ORNL. Successful prosecution of these tests will provide much of the basic information needed for the thorium oxide clustered-pin fuel element.

3.3.3.3. Technical Approach

Since it is expected that the major engineering development will be done under the AI-CE program, the types of tests are only outlined below:

1. Collapse Tests

Problem: Collapse strength at prolonged elevated temperatures.

Objective: Short-time and creep collapse data to provide optimum thickness of cladding.

Course of action: Short-time tests, long-time tests, analytical correlation, elevated-temperature irradiated and unirradiated material properties.

2. Burst Tests

Problem: Internal pressure buildup.

Objective: Burst data to provide optimum thickness of cladding.

Course of action: Short-time tests, long-time tests at elevated temperature, irradiated and unirradiated material properties.

3. General Tests

Problems: Creep, corrosion, fretting, and wear of the cladding.

Objective: Investigation of these problems to provide design information for the full-scale flow test element and the production element.

Course of action: Outpile tests (on a component basis, as required, to provide design information) in a hot organic loop for long duration.

4. Hydraulic and Vibration Test

Problems: Flow distribution, pressure drop, and vibration.

Objective: Hydraulic and vibration characteristics.

Course of action: Outpile flow test of full-sized interim fuel element at low temperature and pressure. Can be performed on a prototypic fuel element before irradiation testing in WR-1.

5. Full-Scale Flow Test

Problems: Flow distribution including inter-channel mixing effects, pressure drop, vibration, and axial loading.

Objective: Hydraulic (including mixing) and vibration characteristics.

Course of action: Outpile flow test of full-sized reference fuel element in a hot organic loop.

It is expected that all of these tests will be performed by AI-CE in developing their UC element and that their information will be directly applicable to the B&W 37-pin oxide element since the same number of pins are used in either element and the pin sizes are approximately equal. The burst testing, however, must take into consideration the higher fission gas pressure buildup in the oxide pin.

Use of a 55-pin thorium oxide fuel element in an HWOCR designed for UC would require separate vibration and flow tests for the reference element. The flow and vibration characteristics of the 55-pin element may differ significantly from those of the 37-pin element.

3.3.3.4. Summary

Engineering tests for SAP-clad clustered-pin fuel elements are being performed under the AI-CE program. B&W will use available data from this program. Additional engineering tests will be performed only to investigate areas peculiar to the thorium fuel element design.

3.3.4. Subtask 25 - Thorium Oxide Fuel Element Assembly Irradiation Tests

3.3.4.1. Objective

The objective of this subtask is to determine the operational behavior of the thorium oxide fuel element assemblies under conditions closely simulating HWO CR conditions.

3.3.4.2. Course of Action

The fuel element assemblies will be irradiated in a succession of experiments. Thus, confidence in the behavior of thorium oxide fuel elements in an HWO CR environment will be progressively developed. The final step will be a proof test in the WR-1 under conditions as closely representative of the HWO CR as reasonably attainable.

3.3.4.3. Status of Technology

The effects of irradiation on thoria-urania fuel are discussed in the description of Subtask 23.

3.3.4.4. Problem Areas

The overall problem is the need to demonstrate that the SAP-clad thorium oxide fuel element design is functionally adequate for operation in an organic coolant under HWO CR operating conditions. Among the items that need verification are (1) mechanical and structural integrity of the cladding and fuel combination during irradiation and (2) the effect of organic fouling during irradiation on the thermal performance of the fuel element.

3.3.4.5. Technical Approach

A number of fuel element assembly irradiation tests will be performed in the WR-1 reactor to obtain information on operational performance at simulated HWO CR conditions. Specific information obtained will include power and temperatures achieved, dimensional stability, extent of fretting wear, gas release, corrosive effects, and other physical and microstructural observations on both fuel and cladding. The largest bundle that can be accommodated, probably a

19-pin cluster, will be irradiated. The pins will be highly instrumented with thermocouples in the fuel and cladding and with flux monitors.

The first irradiation experiment in the WR-1 reactor will operate at a relatively low power level. The second irradiation experiment will operate at a higher power level. The third irradiation experiment will have a greatly improved design because it will be based on the experience of the first experiment and possibly the second experiment.

Additional irradiation experiments in the WR-1 reactor will continue to test the operational characteristics of a reference design fuel element at successively higher power levels until a practical upper limit is achieved. Information obtained from prior irradiations will be incorporated in these experiments. The results obtained will contribute to the design of the 1000 MWe HWOCR thorium dioxide fuel element. As the need for additional irradiation experiments is indicated, they will be further described in amendments to this document. It is presently not feasible to completely predict needs arising from future experimental work.

3.3.4.6. Summary

An irradiation program is planned to determine the operational behavior of the reference thorium oxide fuel element assembly under conditions closely simulating HWOCR conditions. A series of irradiation experiments are planned for the WR-1 reactor. They will start with initial irradiations of a cluster of oxide elements of current design operating at reduced power and culminate in the irradiation of a fuel element assembly at operating conditions permitting a simulated performance of the reference design. The results obtained will contribute to the design of the 1000 MWe HWOCR thorium fuel element.

3.4. Thorium Monocarbide Fuel Element Test Program

3.4.1. Subtask 22 - Thorium Monocarbide Fuel Element Fabrication Development

3.4.1.1. Objective

The objective of this subtask is to establish the 1000 MWe HWOCR thorium-uranium monocarbide fuel composition

and to develop the necessary fabrication and quality control processes.

3. 4. 1. 2. Course of Action

Fabrication methods suitable for a commercial production facility will be developed. Efforts will be directed toward obtaining a fuel composition resistant to attack by moisture.

3. 4. 1. 3. Status of Technology

As will be discussed in detail in the description of Subtask 23, thorium monocarbide is highly reactive with the moisture in air, rapidly hydrolyzing to form hydrocarbon gases and thoria. Abraham of Atomics International⁷⁹ found that 1% zirconium had a definite stabilizing influence on (Th₇U)C, rendering the material capable of being handled in air to some extent. Other stabilizers, such as molybdenum and niobium (columbium) were not as effective as zirconium was. Silicon presented some casting difficulties and was not thoroughly evaluated at AI; however, Battelle⁸⁰ found that silicon was an effective stabilizing addition. Abraham prepared his compound by fusing the elemental components by arc melting with a nonconsumable graphite electrode. Drop casting was abandoned in favor of pouring from a skull-melt furnace since drop casting produced fuel slugs containing voids. The tilt-pour procedure produced sound castings having acceptable physical properties. The melt cycle for a 1000-gram charge was reported to require 2 to 4 minutes at 2000 to 3000 amperes and 32 to 38 volts under a static argon atmosphere at a pressure of 500 mm Hg.

In the BMI investigation,⁸⁰ arc melting of the specimen material on a copper hearth was followed by drop casting of the specimens into a graphite mold. Satisfactory fuel slugs were so obtained.

At Brookhaven (BNL)^{81, 82} thorium monocarbide specimens were prepared by first reacting elemental thorium and carbon in a vacuum for 2 hours at 1800 C (3270 F), crushing to a powder, and then hot pressing the crushed powder in Al₂O₃ dies at 5000 psi at 1300 C (2370 F) in a vacuum. Material having a C/Th ratio of 0.74 was at 95% of theoretical density after hot pressing, and the density increased to 97% after annealing at 1800 C (3270 F) for 10 hours. However, material having a C/Th ratio of 0.96 was at only 80% of theoretical density.

after hot pressing. The density increased after annealing to 90%, but cracking of the pellets was observed.

Vitro Laboratories^{83, 84} has a proprietary plasma-arc melting process for the manufacture of thorium monocarbide powder. The powder is in the shape of spherules, from 1/4-inch diameter to fines (particles finer than 40 mesh are perfect spheres and 100% dense). The larger particles are slightly flattened and contain some central porosity. The powder is recommended for vibratory compaction; however, if pellets are required, the recommended consolidation procedure is arc casting. Conventional pressing and sintering, according to Vitro, would not yield acceptable densities without the presence of free thorium.

Uranium monocarbide technology⁸⁵ may be drawn upon as a guideline for thorium monocarbide fabrication development; however, this must be done cautiously and with the full realization that the two materials are not similar in all respects.

3.4.1.4. Problem Areas

The principal difficulty in fabricating thorium monocarbide is its reactivity with moisture. Because of this reactive tendency, thorium monocarbide must be stabilized by alloying, or dry-processing techniques must be used. After the choice between stabilization and dry processing, casting, finishing, and inspection procedures will be established.

3.4.1.5. Technical Approach

It will be necessary to control the length and diametral dimensions, chemical composition, impurity content, density, and physical intactness to reasonably strict tolerances. The first phase of this investigation will be the development of a relatively stable thorium monocarbide fuel. Stabilizing amounts will be added to the monocarbide in varying proportions, and the pellets will be tested for hydrolysis in air. Although there are a number of possible stabilizing elements, initial work will be restricted to zirconium on the basis of previous success. If zirconium does not acceptably stabilize $(\text{ThU}_{0.02})\text{C}$, other elements or combinations of elements will be considered. A number of small pellets having 2, 5, and 10% zirconium will be arc cast. The

cumulative weight gain in a controlled moisture content environment will then be determined as a function of exposure time. General observations will be made of the condition of the specimens. If an acceptable composition is found, casting, finishing, and inspection techniques will be developed; if an atmospherically stable composition is not found, dry-box processing procedures will be investigated.

Production methods suitable for full-core lots will be developed. The methods will be compatible with recycle and remote fabrication procedures and will produce material of controlled quality.

3.4.1.6. Summary

An experimental program is planned for the fabrication procedures to obtain thorium monocarbide based fuel pellets. The principal difficulty is reactivity with the moisture in normal atmospheric air; this will be overcome by either chemical stabilization or isolated handling.

3.4.2. Subtask 23 - Thorium Monocarbide Fuel Characteristics

3.4.2.1. Objective

The objective of this subtask is to characterize thorium-uranium monocarbide fuel by inreactor and outreactor testing.

3.4.2.2. Course of Action

Necessary fuel materials properties that are not adequately known will be determined or verified under conditions applicable to the HWOCR. The SAP cladding material will be investigated by AI-CE, except for aspects specifically related to thorium fuel. The reference fuel composition will be determined in Subtask 22.

3.4.2.3. Status of Technology

Phase Equilibria

The thorium-carbon equilibrium phase diagram has not been indisputably established; however, the monocarbide region has been sufficiently well defined for the purposes of this subtask. From approximately 45 to 50 atomic % (4 to 5 wt %) carbon, the diagram shows thorium monocarbide to be a single phase (perhaps alpha thorium) existing

without allotropic change to melting at approximately 2625 C (4750 F).^{86, 87} The crystal structure is cubic NaCl.⁸⁸ The thorium-uranium-carbon equilibrium phase diagram has been determined at selected temperatures. Benesovsky and Rudy,⁸⁹ Ivanov and Alekseeva,⁹⁰ and Rosen⁹¹ obtained isothermal diagrams for 1700 and 1000 C (3090 and 1830 F), 1000 and 700 C (1830 and 1290 F), and 950 and 400 C (1740 and 750 F), respectively. There was substantial agreement that the monocarbides of thorium and uranium form a continuous series of solid solutions.

Corrosion and Compatibility

There are three considerations here:

1. Corrosion of the monocarbide by reaction with moisture in the air.
2. Compatibility of the monocarbide with organic coolant.
3. Compatibility of the monocarbide with SAP cladding.

These will each be discussed in turn.

Thorium monocarbide reacts with moisture in the air by hydrolysis. The reaction is fairly rapid, even at room temperature, forming hydrocarbon gases and thorium dioxide. The thorium dioxide exfoliates, leading to complete disintegration of the monocarbide specimen.⁹² The presence of moisture is imperative for this reaction as shown by Farkas et al. of BMI.^{80, 93} They also performed oxidation tests in dry air at temperatures from 480 C (900 F) to 705 C (1300 F), obtaining measurable weight gains after some hours of exposure. Chemical aspects of the hydrolysis reaction were studied by Kempter and Krikorian of LASL.⁹⁴ Abraham of AI⁷⁹ found that addition of approximately 1% zirconium to (Th₇U)C stabilized the monocarbide sufficiently to allow it to be processed in air; it still showed approximately eight times the weight gain of uranium monocarbide.

The investigation at BMI^{80, 93} also included exposure of thorium monocarbide and thorium monocarbide alloyed with 10% uranium monocarbide to organic coolant (Santowax R) at 350 C (660 F). Some slight reaction was found; however, this was not related to the moisture content of the coolant, which was not measured. Additional investigation of this reaction is required.

The BMI investigation^{80, 93} also encompassed the compatibility of ThC and (Th₉U)C with aluminum (specifically, 3003 alloy). No reaction was found after exposure at 540 C (1005 F) for 1000 hours in a helium atmosphere. By analogy with uranium monocarbide,⁶⁹ it might be postulated that the presence of Al₂O₃ on the surface of the aluminum prevents the formation of thorium aluminides and that the alumina is reduced at sufficiently high temperature, allowing the aluminide formation to proceed. As sufficient thermodynamic and kinetic data do not exist to evaluate this further, experimental studies are required.

Physical and Mechanical Properties

Thermal conductivity of thorium monocarbide was reported by Griesenauer et al. of BMI⁹³ and Marchal and Trouve of CEN.⁹⁵ The BMI investigation on (Th₉U)C yielded values varying linearly from 0.0308 cal/sec-cm-C (7.447 Btu/hr-ft-F) at 200 C (390 F) to 0.0442 cal/sec-cm-C (10.688 Btu/hr-ft-F) at 1000 C (1830 F). The CEN work on pure ThC, however, produced much lower values, 0.02 cal/sec-cm-C (4.8 Btu/hr-ft-F) at 160 C (320 F) and 0.016 cal/sec-cm-C (3.9 Btu/hr-ft-F) at 365 C (690 F), which are extremely close to corresponding values for UO₂. Further determinations at higher temperatures and with more refined techniques are required.

The thermal expansion of (Th₉U)C was studied by Griesenauer et al. of BMI,⁹³ and the thermal expansion of (Th₇U + Zr)C was studied by Abraham of AI.⁷⁹ The BMI study extended to approximately 1000 C (1830 F), and the AI work went to 1315 C (2400 F). The studies are in substantial agreement, showing thermal expansion that is approximately linear and results in a 0.7% increase in length at 1000 C (1830 F). Street and Waters of AERE⁹⁶ found that the mean linear thermal expansion of pure ThC was 6.53×10^{-6} /degree C (3.62×10^{-6} /degrcc F) between room temperature and 841 C (1545 F), which is in fair agreement with the BMI value of 7.2×10^{-6} /degree C (4.0×10^{-6} /degree F)⁹⁷ for (Th₉U)C.

Mechanical strength was determined at BMI and AI. The BMI study⁹³ found that compressive fracture stress decreases from 123 ksi at room temperature to 67 ksi at 980 C (1800 F). The AI study⁷⁹ found that tensile fracture stress was 15 ksi at room temperature.

As before, the BMI work was with $(\text{Th}_9\text{U})\text{C}$, and the AI work was with $(\text{Th}_7\text{U})\text{C} + 1\%$ zirconium. BMI also qualitatively determined thermal shock resistance by quenching in oil from 1000 C (1830 F); no evidence of cracking was found. This suggests that the compounds possess a reasonable combination of thermal conductivity, thermal expansion, and strength characteristics, upon which thermal-shock resistance is dependent. The AI investigation also determined that Young's Modulus at room temperature was approximately 14×10^6 psi.

Irradiation Characteristics

A review of the technical literature has not revealed any information on the behavior of thorium monocarbide under neutron irradiation. A complete irradiation program is required to determine such properties as dimensional stability and gas release. Guidelines may be drawn from the irradiation experience with plutonium monocarbide and uranium monocarbide, but this must be done with the full realization that thorium monocarbide, although similar in many respects, is a different material, which may perform differently. In general, studies at ANL,⁹⁸ AERE,⁹⁹ UNC,¹⁰⁰ and other laboratories have shown that arc-fused monocarbides possess excellent radiation stability.

Plutonium Compatibility

The use of plutonium monocarbide in place of uranium monocarbide as the fissionable material will probably not introduce new problems. Admittedly, there is very little data on $(\text{Th}, \text{Pu})\text{C}$; however, the monocarbide of thorium and plutonium is isomorphous and otherwise similar to UC physically and chemically. Nevitt of ANL¹⁰¹ found that the lattice parameters of UC in ThC and PuC in ThC were extremely similar and that both closely followed Vegard's Law. It would certainly be expected that the quaternary situation, $(\text{Pu}, \text{Th}, \text{U})\text{C}$, for the concentrations here specified would not deviate markedly from the previously discussed ternary form, $(\text{Th}, \text{U})\text{C}$.

3.4.2.4. Problem Areas

The principal need is for a determination of the effects of thermal neutron irradiation. Dimensional stability,

chemical stability, and gas release must be determined. The compatibility of fuel and cladding and fuel and coolant must certainly be established. Thermal conductivity must also be positively determined at the operating temperatures.

3. 4. 2. 5. Technical Approach

Thermal Conductivity

Outreactor thermal conductivity will be determined on cast pellet material to at least 2000 C (3630 F). The conductivity will be determined from thermal diffusivity measurements. Experimental determinations will be performed on four compositions having variations in carbon and uranium content of the reference material.

Some of the lower temperature irradiation experiments will be instrumented with thermocouples to determine the inreactor thermal conductivity and to determine the long-term effect of irradiation upon thermal conductivity.

Compatibility

The compatibility study will be divided into three independent areas of investigation as follows:

1. Reaction with moisture in the air. This will be determined in Subtask 22 as part of the fabrication development.

2. Compatibility of the monocarbide with organic coolant. Specimens of the reference fuel material will be exposed in an outreactor loop to flowing organic coolant at temperatures of 260, 370, and 480 C (500, 700, and 900 F) for times of 100 and 1000 hours. Reference organic coolant containing low, nominal, and high moisture content will be utilized. Subsequent analysis will include metallographic examination and determination of weight change data.

3. Compatibility of the monocarbide with SAP cladding. The compatibility between stoichiometric and hypostoichiometric fuel material and SAP cladding material will be determined using contact-loaded diffusion couples. Both bare and anodized SAP surfaces will be tested; investigations will be conducted at temperatures of 425, 540, 650, and 760 C (800, 1000, 1200, and 1400 F) and times of 100, 1000,

and 10,000 hours. The variables will be statistically arranged to minimize testing.

Irradiation Performance

A complete irradiation program is required to properly assess the performance of thorium monocarbide for HWOCR applications. Dimensional stability including swelling as a function of burnup and fracturing due to thermal stress, fission gas release, and compatibility of fuel and cladding under neutron irradiation are the areas of principal importance.

Dimensional stability will be determined by a series of capsule irradiation tests. Thorium monocarbide pellets, some of which will be clad in SAP, will be encapsulated and irradiated to the following burnups: 2500, 5000, 10,000, 20,000, and 25,000 MWD/MTM. These burnups would occur in approximately 1/2, 1, 2, 4, and 5 years of irradiation respectively in a typical test reactor thermal neutron flux. The fuel pins will have varying diametral gap and cladding thickness of 10, 20, and 25 mils to determine the effect of cladding restraint on irradiation stability.

Gas Release

Gas release experiments will produce data from postirradiation sampling to develop a "gas release versus heat rating $\left(\int_{T_1}^{T_2} \lambda d\theta \right)$ " relationship. Experimental parameters will be selected to vary the operating heat rating up to that required to approach center melting as closely as thermal conditions allow. The capsules will be irradiated for the time necessary to accumulate sufficient gas for reliable analysis. Capsule instrumentation will consist of thermocouples and flux monitors where practicable. In addition, the fuel pins irradiated in the dimensional stability subtask will be analyzed for gas release. Dynamic gas release experiments are not planned at this time. However, if there is a future indication that this information is needed, dynamic gas release experiments will be performed.

Compatibility of Fuel With Cladding

A number of the specimens in the dimensional stability and gas release experiments will be SAP clad. Thus, compatibility

between SAP and thorium monocarbide will be determined under neutron irradiation.

3.4.2.6. Summary

An experimental program to characterize the reference fuel is planned. The following investigations are proposed:

1. The thermal conductivity will be determined for temperatures to 2000 C (3630 F). The effect of compositional variations will be considered.
2. The compatibility between fuel and organic coolant will be surveyed at various temperatures and moisture contents.
3. The compatibility between fuel and cladding will be determined using contact-loaded diffusion couples. Experimental variables will include carbon content and time and temperature of exposure.
4. Capsules of the reference fuel will be irradiated to determine dimensional stability, fission gas release, and the effect of neutron irradiation upon thermal conductivity and compatibility with SAP cladding. Burnups will be from 2500 to 25,000 MWD/MTM.

3.4.3. Subtask 24 - Thorium Monocarbide Fuel Element Engineering Tests

The objective of this subtask is to perform outpile engineering tests on the thorium monocarbide clustered-pin fuel element to obtain additional design and performance data to ensure the structural adequacy of the fuel element for operation in an HWOCR.

The same tests required for the 37-pin thorium oxide element will be needed for the 37-pin thorium carbide element (see section 3.3.3). AI-CE is expected to perform these tests in developing the 37-pin UC element, and the results will be directly applicable to the B&W carbide element since the number of pins is the same and the pin sizes are approximately equal.

3.4.4. Subtask 25 - Thorium Monocarbide Fuel Element Assembly Irradiation Tests

3.4.4.1. Objective

The objective of this subtask is to determine the operational behavior of the thorium monocarbide fuel element

assemblies under conditions closely simulating HWOOCR conditions.

3.4.4.2. Course of Action

The reference fuel element assemblies will be irradiated in a succession of experiments. The final step will be a proof test in the WR-1 under conditions closely representative of HWOOCR conditions

3.4.4.3. Status of Technology

There is no irradiation experience with thorium monocarbide.

3.4.4.4. Problem Areas

The major problem is the need to demonstrate that the SAP-clad thorium monocarbide fuel element design is functionally adequate for operation in an organic coolant at HWOOCR operating conditions. Among the items that need verification are (1) dimensional stability of the fuel element during irradiation, (2) mechanical and structural integrity of the cladding and fuel combination during irradiation, and (3) the effect of organic fouling during irradiation on the thermal performance of the fuel element.

3.4.4.5. Technical Approach

A number of fuel element assembly irradiation tests will be performed in the WR-1 reactor to obtain information on operational performance at simulated HWOOCR conditions. Specific information obtained will include power and temperatures achieved, dimensional stability, extent of fretting wear, fission gas release, corrosive effects, and other physical and microstructural observations on both fuel and cladding. As large a bundle as can be accommodated, probably a 19-pin cluster, will be irradiated. The pins will be instrumented with flux monitors and with thermocouples in the fuel and cladding.

The first irradiation experiment in the WR-1 reactor will operate at a relatively low power level. The second irradiation experiment will operate at a higher power level. The third irradiation experiment will be of improved design based on the experience of the first experiment and possibly the second experiment.

Additional irradiation experiments in the WR-1 reactor will continue to test the operational characteristics of the fuel element at successively higher power levels until a practical upper limit is achieved. Information obtained from prior irradiations will be incorporated in these experiments. The results obtained will contribute to the design of the 1000 MWe HWO CR thorium fuel element. As the need for additional irradiation experiments is indicated, they will be further described in amendments to this document. It is presently not feasible to completely predict needs arising from future experimental work.

3.4.4.6. Summary

An irradiation program is planned to determine the operational behavior of the thorium monocarbide fuel element assembly under conditions closely simulating HWO CR conditions. A series of irradiation experiments are planned to be performed in the WR-1 reactor. They will start with initial irradiations of a cluster of thorium monocarbide elements of current design operating at reduced power and will culminate in the irradiation of a fuel element assembly at operating conditions which simulate the performance of the reference design. The results obtained will contribute to the design of the reference 1000 MWe HWO CR thorium fuel element.

3.5. Additional R&D Required for Commercial Acceptance

Sections 3.2, 3.3, and 3.4 outline the test programs required to demonstrate the performance of prototypic thorium fuel elements in an HWO CR environment. The fuel element test program is the principal development task; however, additional R&D will be required to achieve commercial acceptance of the HWO CR thorium fuel cycle. Experimental physics data are required for confident design of a Thorium HWO CR. Thorium-U-233 recycle must be converted to practicality on technical and economic bases. A discussion of the R&D in these two areas follows.

3.5.1. Experimental Physics

The experimental physics program required to support the development of the thorium cycle for the HWO CR is based on an analysis of the nuclear design problems associated with that concept

and an evaluation of the applicability of available information. The design features of the Thorium HWOCR's are described in section 2 of this report, and the available data have been reviewed before.^{101, 102} Additional information on the thorium cycle has been obtained from experimental programs in progress at the Savannah River Laboratory (SRL) and Brookhaven National Laboratory (BNL), as will be discussed below.

The total experimental program can be separated into two phases. The first phase consists of basic experiments needed to develop analytical methods, substantiate the crucial features of the reference design, and provide a firm basis for evaluating the merits of the concept. The second phase is more design-oriented and includes those experiments needed to confirm the design and operational characteristics of the reference reactor. The experiments required for each phase are outlined in the following sections, although it must be recognized that additional work may be needed as the problems become better defined through further design and development studies. It should also be pointed out that there is no clean break between the phases. Depending on the development schedule, funding, incentives, and other factors, the sequence of experiments may be altered.

The program is based on the assumption that the demonstration of the Thorium HWOCR be initially with U-233 or U-235. A decision to fuel the reactor with plutonium would require major revisions in the development program. It is also assumed that the physics experiments will be performed at SRL, although this work could be done at B&W.

3.5.1.1. Phase I

The objective of Phase I is to provide the integral physics data needed to confirm or normalize the analytical methods and cross sections that are used in the most crucial design calculations. The program incentives are based on the predictions of power cost and conversion ratio. However, as discussed in section 4 the error band in the economic evaluations resulting from uncertainties in the nuclear calculations alone is sufficiently broad that some experimental confirmation of their reliability—or narrowing of the error band—is absolutely

essential at an early stage. Accurate cross sections are needed, but the principal problem is the reliability of the analytical model used to treat the complex and dissimilar geometrical features of close-packed rods and tubes. The conversion ratio and power distribution are particularly sensitive to the homogenization model used.

Phase I should also include any experiments needed to confirm the technical feasibility of the concept. However, at this stage in the design and development program, no go or no-go items have been identified. There may be some fundamental questions on safety that should be included in Phase I, but this decision must await further evaluation.

The experiments required in Phase I are outlined below. They should be regarded as minimum requirements; earlier information on some of the problems relegated to Phase II would be desirable, but probably not absolutely essential.

Task 1: Basic Integral Data

Two sets of integral experiments (both exponentials) are essentially complete—one at SRL and one at BNL. SRL measured parameters for U-235-ThO₂ clusters and U-235-thorium metal nested cylinders. BNL has done a short but pertinent series of measurements with U-233-ThO₂ clusters. Data from these two programs are limited, but will provide the first real tests of the physics models. Considerably more experimental work is needed, however, if a high degree of confidence in the analytical calculations for these systems is to be achieved.

The broad objective of Task 1 is to obtain basic integral cross section data needed to satisfy the objectives of Phase I. For the case of U-233, the data are obtained more easily and are more directly applicable if they are measured in lattice-type experiments, as will be discussed under Task 3. For Th-232, there are major uncertainties in both the thermal (non-1/v) and epithermal cross sections. The low energy range should be covered by detailed microscopic cross section measurements. Although such measurements are beyond the scope of the program proposed here, other laboratories (e. g. , BNL) should be encouraged to undertake this work.

The principal integral cross section to be measured in this task is the effective resonance integral of thorium. Since this parameter is geometry-dependent, effective resonance integral measurements are needed to confirm both the resonance parameters and the complexities of the analytical model. The case of clustered pins of ThO_2 is fairly well established from direct resonance integral measurements as well as the SRL lattice experiments. Although no data are available for clustered pins of ThC , the nuclear properties are similar enough to those of ThO_2 pin clusters that the risk in omitting such measurements, in Phase I at least, is probably justifiable.

However, the geometrical differences for the case of the nested metal tube fuel element are sufficiently large that, in the absence of any other confirmatory data, direct measurements of the effective resonance integral are necessary. The measurements can be made using well-established activation techniques and should cover a range of tube diameter, wall thickness, and number of tubes appropriate to the conceptual design of this fuel element.

The thermal scattering parameters for Santowax and Dowtherm (simulated coolant in exponential experiments) must be measured so that scattering kernels, required as input to computer programs such as THERMOS, can be derived. THERMOS or codes of equivalent sophistication are required for adequate calculations of effective thermal cross sections, flux and power distributions through the fuel elements, temperature coefficients, and other thermal phenomena.

Task 2: Fuel Element Model Test

Integral data are needed to test the theoretical models used to treat the geometrical features of the three alternate fuel element designs, so that the intercomparisons used to select the reference design are done on a firm basis. For the case of clustered ThO_2 pins, the exponential experiments done at SRL should be adequate for the preliminary model test needed in Phase I, unless the analysis of results indicates serious discrepancies. The case of clustered ThC pins is probably not sufficiently different to require a separate preliminary model test. However, for the case of nested thorium metal tubes,

the experiments at SRL do not appear to be sufficient to provide an equally strong test of the analytical methods, and additional work appears to be appropriate.

Experiments should be performed with nested tubes of U-235-thorium alloy, having dimensions and enrichments appropriate to the reactor design. To minimize fuel requirements, the use of the PCSTR-technique is recommended, and the fuel elements fabricated for SRL can probably be used. The measurements should be made as a function of lattice pitch and number of tubes per nest; in most cases the coolant should be organic, but H_2O , D_2O , and air should also be used to provide additional checks on the analysis. The following are typical examples of the measurements that should be made:

1. Infinite medium multiplication factor (k_∞).
2. Epithermal-to-thermal captures in Th-232 (ρ_{28}).
3. Epithermal-to-thermal fissions in U-235 (δ_{25}).
4. Epithermal-to-thermal fissions in U-233 (δ_{23}).
5. Modified conversion ratio (MCR).
6. Thermal disadvantage factor ($\bar{\phi}_m/\bar{\phi}_f$).
7. Power distribution [$P(\bar{r})$].
8. Thermal and epithermal neutron spectra [$\phi(E)$].

In some cases, the spatial distribution of these parameters should be measured.

Task 3: U-233 Experiments

Experiments on U-233-fueled lattices are needed in Phase I to ensure the reliability of fuel cycle cost and conversion ratio calculations. The exponential experiments in progress at BNL do not appear to be sufficient for this purpose, so additional experiments are recommended. By studying the U-235-Th- D_2O system comprehensively and obtaining model and cross section tests separately (Tasks 1 and 2), the effect of U-233 can be isolated to a large extent, and U-233 experiments can be minimized. The experiments proposed parallel those of Task 2, except that the fuel contains about 2% U-233. To obtain a direct comparison between U-235 and U-233, the same range of parameters should be covered, and all of the measurements listed under Task 2 should be made. However, those

measurements providing more insight into the neutron spectrum, effective η , and conversion ratio should be emphasized. The coolant void coefficient should also be considered in more detail because of its role in nuclear safety.

3.5.1.2. Phase II

The objective of Phase II is to confirm the detailed design of the reference reactor and to determine its operational characteristics. At this stage, the range of variables (fuel element dimensions, lattice pitch, etc.) will have been narrowed. The experimental program should explore this range of variables.

Task 4: Burnup Studies

Data are needed to confirm design calculations of the core lifetime and integrated conversion ratio because of the importance of these parameters in the economic evaluation of the concept. The effect of fission products on the reactivity and conversion ratio and the changes in heavy-element composition (particularly Pa-233) should be measured by irradiating samples of U-233-thorium and U-235-thorium fuel pins to 10,000 to 20,000 MWD/T. The reactivity changes can be measured periodically using an RMF-type of facility,¹⁰³ in which the neutron-producing and neutron-absorbing components can be separated. The fuel conversion ratio should also be measured by isotopic analysis.

Task 5: Operating Coefficients

The operating coefficients—moderator, coolant, and fuel temperature coefficients of reactivity—can be measured in a number of ways. The least complex way is to measure them in a zoned critical assembly or PCTR, using a heated central region. The moderator temperature coefficient can be measured by heating D₂O to approximately 80 C and observing the reactivity change. The same approach can be taken to obtain the coolant temperature coefficient, except that the temperature range is much higher and the fuel element must be insulated from the moderator. These experiments should be supplemented with total-coolant voiding measurements. The fuel, or Doppler, coefficient can also be measured in the same facility, using relatively standard activation or reactivity techniques.

Task 6: Critical Assembly Mockup

There are a number of important nuclear properties of the reference reactor that cannot be determined adequately in the small-scale experiments described above; therefore a relatively large critical assembly will be necessary to measure them. It should not be necessary to mock up the reference reactor in full size and exact detail; a large zoned system may be adequate. However, the degree of simulation cannot be specified until the reference HWOCR design is established and the performance requirements for the first thorium core are defined. It is assumed that the thorium core will be loaded in an existing HWOCR.

The following are typical examples of the measurements that should be made in the mockup assembly:

1. Power distribution. Measure the gross and local power distribution and power sharing if radial or axial zone loading is used.
2. Operating coefficients. Confirm the earlier measurements of moderator, coolant, and fuel reactivity coefficients.
3. Kinetics. Determine the zero-power transfer function. Develop experimental methods of studying stability, xenon oscillations, etc.

Task 7: Startup and Operational Tests

Although this task is beyond the scope of the development program, it is mentioned for completeness. If a fresh thorium core is loaded, the usual startup tests, safety measurements, and periodic checks of the nuclear parameters throughout core life will be made. However, if thorium fuel elements are loaded by on-line refueling methods, the principal measurements will be aimed at determining the perturbations introduced by the thorium fuel in the UC core.

3.5.2. Recycle Technology

The flow sheets selected for the recycling of thorium-uranium alloy and thorium-uranium oxide fuels are presented in BAW-393-3 (section 2).¹⁰⁴ A flow sheet for the recycling of thorium-uranium carbide has not previously been presented. However, basic assumptions concerning the problem areas in the processing of all three fuel types

in a TRIP (Thorium Recycle Integrated Plant) facility are discussed in this section.

The major philosophy governing the TRIP facility flow sheet and process equipment selection was the use of established re-processing, conversion, and fabrication techniques. This does not require that each technique be proven with the specific fuel for which it is chosen in order to merit consideration. This is not possible with the present state of thorium technology. However, novel concepts, no matter how promising they may appear to be, are excluded from consideration unless they have been previously applied successfully on an industrial scale. Thus, the flow sheets which evolved for the alloy and the oxide fuels, though somewhat unique, are a balance between optimum economics and minimum risk. Where possible, the selection of the carbide fuel processes is handled in an identical manner.

In order for the TRIP recycle concept to attain commercial acceptance, a considerable amount of development work is required. A study has been made to determine the areas requiring additional research and development. As a result the problem areas (to permit commercial acceptance of the TRIP design) have been arranged into two categories:

1. Those chemical and metallurgical problems which must be solved to make the TRIP facility operable as conceptually envisioned in BAW-393-3 for oxide and metal (similarly for carbide).
2. Those problems which need to be solved in order to design a TRIP facility today based on BAW-393-3 flow sheets employed current state-of-design technology.

Those problem areas related to category 1 are listed in Table 3-1, and those problem areas related to category 2 are listed in Table 3-2.

3.6. Summary

Selection of a reference fuel element for future development will be based primarily on the following factors: economics, fuel utilization, compatibility with AI-CE design (including R&D spinoff), and technical feasibility (including development costs and schedule). The preceding discussion of R&D requirements, therefore, provides important input

and background for the comparative evaluation presented in section 4. The test program plan for the selected fuel element will be used as the starting point for preparing a detailed work program to be submitted to the Commission.

Table 3-1. Chemical and Metallurgical Uncertainties in Thorium-Uranium Fuel Processing

Item	Fuel type		
	ThO ₂ -UO ₂ (XAP clad)	Th-U alloy (Zr clad)	ThC-UC (XAP clad)
Choice of dissolution operation (e. g. , chemical decladding versus simultaneous dissolution)	X	X	X
Effect of cladding dispersion material (Al ₂ O ₃) on dissolution cycle	X		X
Effect of organic degradation films on dissolution cycle	X	X	X
Practicality and sequence of heel type dissolution with HWOCR fuel	X		
Effect of fluoride ion concentration on feed adjustment		X	
Other feed adjustment requirements	X	X	X
Off-gas problems during core dissolution			X
Effect of soluble hydrolysis products on extraction system operation			X
Complications produced by the addition of zirconium to the fuel mixture		X	X
Choice of solvent extraction flow sheet	X	X	X
Minimum extraction requirements for remote recycling of fuel	X	X	X
Nitration reaction during product concentration	X	X	X
Acceptability of steam denitrated thoria as an intermediate reactant		X	X
Pyrophoricity or reactivity of intermediate or final fuel material		X	X
Atmospheric requirements in processing steps		X	X
Choice of fuel preparation method			X
Ability to meet purity specification		X	
Fuel homogeneity		X	
Fuel stabilization			X
Interaction of fuel and cladding		X	

Table 3-2. Uncertainties in Design of Thorium-Uranium
Fuel Processing Equipment

Item	Fuel type		
	ThO ₂ -UO ₂ (XAP clad)	Th-U alloy (Zr clad)	ThC-UC (XAP clad)
Proper dissolver design for the accommodation of desired operation sequence and criticality safety	X	X	X
Fuel element disassembly procedure	X	X	X
Choice of type of contactors to use in extraction plant	X	X	X
Adaptation of feed preparation steps to continuous operation	X	X	X
Adaptation of feed preparation steps to remote operation	X	X	X
Methods of reducing sorbed gases in fuel	X		X
Fuel fabrication method			X
Satisfactory joining techniques for cladding	X		X
Adaptation of element fabrication to remote operation		X	

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- ¹⁰³ DeBoisblanc, D. R., et al., Second UN International Conference on Peaceful Uses of Atomic Energy, A/Conf 15/P/1842, Vol. 13, p 75, Geneva, 1958.
- ¹⁰⁴ Moncrief, E. C. and Schileo, G., Recycle of HWOCR Thorium Fuel, The Babcock & Wilcox Company, BAW-393-3, Lynchburg, Virginia, March 1966.

4. COMPARATIVE EVALUATION OF THORIUM FUELS

Section 2 describes fuel element designs for the thorium fuel forms under consideration, and section 3 outlines test programs for developing the fuel elements. With this background established, section 4 summarizes the characteristics of each design in a head-on comparison with respect to the principal evaluation factors.

4.1. Economics

4.1.1. Ground Rules and Methods

4.1.1.1. Economic Ground Rules

The economic analyses presented here are based on two representative methods of utility financing: investor-owned and publicly-owned reactors. The principal economic factors for these methods of financing are as follows:

	<u>Investor- owned</u>	<u>Publicly- owned</u>
Fixed charges on plant capital, %/yr	12	7
Fixed charges on non-depreciated assets, including D ₂ O and fuel investment, %/yr	10	5
Present worth discount rate, %/yr	6	4
Annual plant factor	0.8	0.8

These economic parameters are representative of one of several 1000-MWe units committed for on-line service in the 1970's; these units are forerunners in a growing HWOCR economy. Accordingly, the economic parameters related to fuel cycle costs are based

on large-scale fabrication and processing geared to service an expanding HWO CR economy. Additional economic ground rules are as follows:

1. The reactor will begin operation in the period 1970-1980.
2. The plant lifetime is 30 years.
3. The thorium system is initially fueled with U-235, and the uranium produced is recycled to the core in the reference cases. Plutonium startup and recycle are also considered.
4. For capital cost estimates, all components are fully developed, and first-of-a-kind engineering has been done. (The Thorium HWO CR will use the AI-CE overall plant design and capital cost data where practical.)
5. Fuel is privately owned.
6. The reactor plant and recycle facilities are operating in a mature industry of 15,000 MWe.
7. The reference values of materials are:

U-233, \$/gm	14
ThO ₂ (nitrate form), \$/lb	5
U ₃ O ₈ yellowcake, \$/lb	8
Separative work, \$/kg	30
D ₂ O, \$/lb	20
Fissile plutonium, \$/gm	10

4. 1. 1. 2. Capital Cost Methods

Since the major effort on the Thorium HWO CR system is concentrated on core designs, the plant design data are insufficient for establishing independent overall capital cost estimates. However, some consideration of plant capital cost variations is required to optimize for economic power, since design changes that reduce one component of the power cost may increase another. Accordingly, reactor hardware cost curves and secondary system heat transfer equipment costs based on data from the ORNL evaluation of the HWO CR¹ were developed as reported in BAW-393-6,² and used to estimate capital costs of the reference designs. Other variable costs, such as primary pump and driver and instrumentation, were included. The effect of design parameters on net plant efficiency was also estimated, and the capital costs for the three designs reflect these variations in efficiency.

4. 1. 1. 3. Fuel Cycle Cost Methods

The fuel cycle economic evaluation considers all costs associated with fuel, from the time the initial fuel loading is purchased for fabrication to the end of reactor lifetime when fuel is discharged from the reactor at some value. The costs include fabrication, reprocessing, shipping, and fuel depletion (direct costs), and working capital charges on investments, either material of value (inventory) or services of value (fabrication and reprocessing).

Economic evaluation of nuclear fuel cycles can be complicated, since many factors vary during the lifetime of a nuclear plant; therefore a method of evaluation that properly accounts for the time value of money is required. Many investor-owned electric utilities in the United States use the present worth of future revenue requirements as the criterion of economic choice. This method, based on minimizing the cost of energy to the customer, can be readily applied to fuel cycle evaluation. The appropriate economic criterion is the levelized fuel cycle cost which is derived by applying the present worth to annual fuel cycle revenue requirements. This method, incorporated into one of the Company's confidential computer programs, has been used to obtain the nuclear fuel costs in this analysis. The levelized fuel costs are defined as follows:

1. Let M = levelized fuel cost (mills/kwhr) that satisfies fuel-cycle revenue requirements during a period of n years.
2. Let K_1 = kwhr generated in the first year, K_2 = kwhr generated in the second year, and so on.
3. Let P_1 = present worth of \$1 received one year hence, P_2 two years hence, etc.
4. Let R_1 = revenue requirements in the first year, R_2 in the second year, etc.
5. Since M , by definition, makes receipts satisfy the revenue requirements, it is given by

$$M = \frac{P_1 R_1 + P_2 R_2 + \dots + P_n R_n}{K_1 P_1 + K_2 P_2 + \dots + K_n P_n} \times 10^3$$

Levelized fuel costs for a 30-year period are calculated for the 1000-MWe Thorium HWO CR conceptual designs based

on physics data from recycle analysis. In the recycle analysis, lifetime calculations are made for a pressure tube unit cell and include a correction for total core leakage as noted in section 2. The fuel exposure history of a fuel element is calculated by batch core techniques modified to simulate continuous on-line refueling by allowing the batch cores to run to a lifetime exactly double the normal lifetime. The basis for this simulation is that the fuel exposure of the discharged fuel is about twice the average exposure in the core. The validity of this method was tested and found to be adequate for survey calculations, but inadequate for establishing an optimum startup cycle. Consequently, most of the fuel cycle costs are not based on the startup core, but on the first fuel element of the second core. This fuel element, which displaces one of the six original elements in a pressure tube, is discharged from the core at the same burnup as the equilibrium cycle fuel burnup. The startup core fabrication cost is included in the fuel cost of the first cycle calculated.

The optimum startup cycle will have a lower average fuel exposure than will the equilibrium cycle; therefore, the fabrication component of the startup cycle fuel cost will be greater than the value calculated by including the startup core in the first-cycle fuel cost analysis. However, the fuel enrichment of the startup core will be lower than that of the first-cycle core—a fact that will compensate for the disparity in fabrication cost. Detailed fuel management calculations have not been made to optimize the startup cycle, but a startup method was considered for the nested-cylinder core, as explained in section 2.3.1, to verify the assumption that fuel cycle costs based on the batch core nuclear analysis approximate those for a true startup method.

Although the components of the fuel cycle cost and the method of computation are adequately established throughout the power industry, the correct working capital charges on fuel inventory in recycled cores are not clearly defined. This is due partially to differences in accounting methods by various utilities and designers and partially to the uncertainty of the worth of bred fuel.

Several methods of computing fissile material inventory charges have been considered. In the reference calculations, a value of \$14/gm is assigned to U-233, and working capital charges on

fissile material are calculated on an average fuel worth during each cycle. Fabrication charges are linearly depreciated to zero value at the end of core life. Reprocessing and shipping charges, which occur at the end of the cycle, are capitalized from the beginning of the cycle and hence are negative costs.

Initial core fabrication financing charges are based on an equivalent fabrication rate of 0.5 tonne/day, an average value during the fabrication period equal to one-half the initial core value and a rate of 6% and 4%, respectively, for investor- and publicly-owned plants. These financing charges are added to the fabrication cost of the first core loading. The pre- and post-irradiation times used in computing the fuel cycle costs are 90 and 210 days.

Fabrication-reprocessing costs are a significant portion of the fuel cycle cost. To minimize the uncertainty in this component, B&W evaluated the fabrication and reprocessing requirements in detail for both thorium metal nested-cylinder and Vipac oxide pin fuel element designs. Cost estimates were developed based on both thorium and uranium recycle in a heavily shielded, remotely operated integrated plant, the TRIP facility, and the results were published in BAW-393-3.⁶

Personnel at ORNL also calculated fabrication and reprocessing costs for the same designs to aid in their design evaluation effort, and provided cost information from a computer program on several other designs to assist B&W in the optimization studies. The fabrication-reprocessing values used in the fuel cycle cost calculations are as follows:

	<u>Nested cylinder</u>	<u>Oxide pin</u>	<u>Carbide pin</u>
Fabrication and conversion	40.30	42.00	51.20
Reprocessing and shipping	27.10	29.00	29.00

The fissile material charges account for the change in fuel value from the initial worth, due to fuel depletion (make-up) and losses in processing. All losses (1.5%) and protactinium decay (98.95%) are assumed to occur at the end of core life.

4. 1. 2. Economics of 1000-MWe Thorium-Optimized HWOCR's

4. 1. 2. 1. Fuel Cycle Costs

Table 4-1 summarizes the fuel cycle costs of all three designs for both investor-owned and publicly-owned utility financing, based on U-235 startup and batch recycle of U-235 and bred U-233. The fuel costs for plutonium startup are given in Table 4-2.

In general, the U-235 startup fuel cycle costs are between 1.19 and 1.23 mills/kwhr for all designs at 20,000 MWD/T under investor-owned utility financing. Under publicly-owned utility financing, the fuel cycle costs vary from 0.94 to 1.00 mill/kwhr. The costs are lowest for the nested-cylinder design and highest for the carbide pin design, but the difference is within the estimated uncertainty band. It is also noteworthy that the fuel cycle costs based on batch recycle and those based on semi-continuous refueling methods (see section 2.3.1) are about equal.

Based on the same thermal performance as the U-235 case, the use of plutonium priced at \$10/gm as startup fuel with recycle of plutonium and bred U-233, gives fuel cycle costs essentially identical to those for U-235 startup fuel at \$12/gm. These results may be altered appreciably when one considers the variation in power production over the life of the fuel element that occurs when high-cross-section plutonium is burned and low-cross-section uranium is produced. At least, a reoptimization of the design or zone loading would be required to compensate for this effect.

4. 1. 2. 2. Capital, D₂O, and Maintenance Costs

The estimated capital, D₂O, and operating costs for each of the three reference designs are summarized in Table 4-3. Capital costs for investor-owned financing conditions are $\$119 \times 10^6$ for the nested-cylinder design and $\$126.4 \times 10^6$ for the oxide pin design. Indirect costs were determined using the same percentages as used in the advanced converter study.³ These costs compare with $\$120.9 \times 10^6$ estimated by ORNL for the uranium design and $\$115.3 \times 10^6$ for the thorium metal design in the HWOCR Evaluation Report.¹ Including the D₂O cost and a constant value of 0.3 mill/kwhr for operation and maintenance,

the power costs, exclusive of the fuel cycle, vary from 2.36 to 2.51 mills/kwhr. Corresponding values under publicly-owned financing conditions are 1.46 mills/kwhr for the nested-cylinder design and 1.55 mills/kwhr for the oxide pin design.

4. 1. 2. 3. Power Costs

The fuel cycle costs have been combined with the other costs to obtain the following total power costs.

<u>U-235 startup fuel</u>	<u>Nested-cylinder core</u>	<u>Oxide pin core</u>	<u>Carbide pin core</u>
Investor financing	3.54	3.73	3.71
Public financing	2.40	2.49	2.52
<u>Plutonium startup fuel</u>			
Investor financing	3.54	3.73	NC*
Public financing	2.40	2.50	NC*

* Not calculated.

No reoptimization of the designs was made for the publicly-financed cases even though the bases of cost were different. Reoptimization of the lower base of cost would result in a core with higher inventory and lower specific-power, a higher conversion ratio, and somewhat lower power costs. Insufficient work has been completed to indicate the changes that would be required to optimize the plutonium fueled cores.

In general, the power costs under investor-owned financing conditions vary from 3.54 mills/kwhr for the nested-cylinder design to 3.73 mills/kwhr for the oxide pin core with U-235 startup fuel. Under public-financing conditions the nested-cylinder power costs are lowest, 2.40 mills/kwhr, and the carbide, rather than the oxide, power costs are highest, 2.52 mills/kwhr.

With plutonium startup fuel, both the absolute and relative values of the power costs are essentially the same as for the U-235 cases. However, as previously noted, some redesign would be required to compensate for the higher peaking factors that occur with

plutonium fuels. Thus, these values serve more to indicate the potential for the use of plutonium in the Thorium HWO CR rather than to provide accurate power costs with this fuel.

4. 1. 3. Fuel Cycle Costs - 1000-MWe AI-CE HWO CR With Thorium Fuel Elements

The preliminary thorium-fuel-element designs described in sections 2. 3. 2 and 2. 4. 2 were developed for operation in a 1000-MWe HWO CR defined by extrapolation of the 500-MWe HWO CR demonstration plant described in AI-CE Memo 25.⁴ Since the optimized thorium carbide core described in section 2.5 is designed to operate in the AI-CE HWO CR plant, two separate carbide designs are not required.

The batch fuel cycle cost based on "idealized" cycle nuclear calculations are shown in Table 4-4 and Figure 4-1 for these three designs, along with corresponding results for the reference Thorium HWO CR designs. Although the optimum and the thorium carbide core for the AI-CE plant are identical, the "idealized" batch fuel cycle costs are presented for comparison with those for the oxide pin and nested-cylinder designs.

As shown in Figure 4-1 the fuel cycle costs for the 55-pin thorium oxide core in the AI-CE plant are about 0.1 mill/kwhr higher than those for the 37-pin optimized thorium oxide design. This difference is primarily due to the design requirements which dictate pin diameters in the 55-pin fuel element of about 0.38 inch, which is below the economic optimum primarily because of the increased fabrication cost in \$/kg. In addition, the resultant lower volume fraction of thorium decreases the conversion ratio and increases the burnup cost to essentially offset the savings from lower fuel inventory. Thus, with oxide fuels, either the diameter or the quantity of pressure tubes must be increased for minimum fuel cycle costs.

As noted in section 2. 3. 2, it was assumed that the pressure tubes in the AI-CE plant that are not required for the thorium metal nested-cylinder core will not be used. Modifications to make the core and plant more compatible while maintaining the advantageous higher-power-capability would definitely be required, but the results illustrate the potential of the nested-cylinder element. Table 4-4 shows that the batch "idealized" fuel cycle costs for the thorium metal nested-cylinder

core in the AI-CE plant are essentially identical to those for the optimized design. Thus, the favorable economics of the thorium-optimized nested-cylinder design is due to reduced capital and D₂O cost rather than to the fuel cycle cost.

4. 1. 4. Major Uncertainties

4. 1. 4. 1. Fabrication and Reprocessing Cost Uncertainties

Unit processing costs were estimated by B&W and ORNL for the thorium-optimized fuel element designs described in section 2 (see Table 4-5). All costs are based on a burnup of 20,000 MWD/T, which corresponds to a recycle plant capacity of about 2.6 tonnes/day.

The ORNL costs were compiled from two sources. The costs for reprocessing, shipping, and conversion are from ORNL's HWO CR evaluation.¹ To these costs were added more recent ORNL costs⁵ for fabricating the optimum fuel designs mentioned above.

The B&W estimates for oxide and alloy were made by adjusting the TRIP facility's⁶ recycle costs for recent changes in the pin and nested-cylinder designs. Data from reference 5 and other sources⁷ were used to adjust these costs.

B&W carbide costs were estimated by adding the ORNL costs for reprocessing, conversion, fabrication, and shipping after making these adjustments:

1. A 10% cost increment was added to the ORNL unit reprocessing cost for oxide to give the carbide recovery cost under the B&W single-cycle concept. This incremental cost to handle carbide was considered valid by SRP personnel.⁸
2. A 20% cost increment was added to the ORNL carbide fabrication cost to cover flow sheet uncertainties (to expedite estimation ORNL assumed the same fabrication flow sheet for ThC as was used previously for UC).
3. The total adjusted unit cost based on the ORNL estimates was then decreased by 10% for integrated processing (credit to the TRIP recycle concept).

The agreement between the estimates in Table 4-5 is extremely good, although perhaps fortuitous. While the maximum difference between estimates is only about 10% (for the alloy fuel), the potential error in the absolute costs may be 20 to 25%. Nuclear fuel cycle cost estimates are seldom performed in sufficient breadth to permit assignment of statistically evaluated error bands. Normally, when several estimates are used to cost the same process, the technical approach and assumptions vary considerably between estimates. In this case, the error band assigned to the cost by each estimator generally reflects his opinion of the error based on his overall knowledge of the problem. Such is the case in the estimates of the unit processing costs given in Table 4-4.

Two contributions should be considered in evaluating the total error associated with B&W's HWOCR unit processing cost estimates: (1) an error associated with the level of effort expended on the design, and (2) an error dependent on the technological unknowns inherent in each fuel recycle flow sheet. These error bands are estimated to be 15% and 10% for the most-proven fuel type (oxide). Greater uncertainties are possible for the metal and carbide cases.

Since unit processing costs constitute about 40% of the HWOCR fuel cycle cost,² a potential error of 25% in the unit processing cost can result in about a 0.1 mill/kwhr uncertainty in the total power cost. Thus, a total power cost difference of 0.1 mill/kwhr between candidate fuels falls within the possible uncertainty band. Therefore the final choice of fuel should be based on careful consideration of other factors in addition to economics.

4.1.4.2. Fuel Exposure Limits

As noted in the evaluation of each design (section 2) the exposure limit of 20,000 MWD/T is not the economic optimum. Figure 4-1 illustrates typical power cost variations with MWD/T and indicates that fuel cycle costs vary about 0.01 mill/kwhr per 1000 MWD/T change in exposure for small changes from the reference value. In general, an increase to the economic optimum of 25,000 to 35,000 MWD/T would decrease fuel cycle cost of each design about 0.05 to 0.10 mill/kwhr. At burnup less than the design value the rate of change in fuel cycle cost

with MWD/T is increasingly greater. A reduction to 10,000 MWD/T would increase the cost 20 to 25%.

The value of 20,000 MWD/T established for the oxide design is limited by fission gas pressure and represents a compromise between two economically important parameters—MWD/T and specific power or pin heat rating. Should development efforts produce favorable fission gas release results, exposures could probably be increased at least 50% before other irradiation characteristics begin to limit performance.

The nested-cylinder exposure of 20,000 MWD/T average is based on maximum fuel swelling of about 7% at the rate of 2 volume % per atomic % burnup. This swelling rate is probably conservative, judging from PNL results. Furthermore, the change in volume at design conditions would not be limiting unless the structural integrity of the fuel element were jeopardized. Since this situation demands engineering judgment, an attempt was made to establish reasonably conservative limits, and favorable development results could demonstrate that economically optimum exposure is achievable.

No irradiation data are available for thorium monocarbide. The design value of 20,000 MWD/T is assumed on the basis of uranium monocarbide data, but cannot otherwise be justified.

4. 1. 4. 3. Conversion Ratio Uncertainty

The conversion ratio in any reactor is primarily determined by two factors: the net number of neutrons produced per absorption in fissile material ($\eta - 1$), and the ability of the fertile material to compete with other neutron-absorbing atoms and losses. Since each of these items has associated with it some degree of uncertainty, the conversion ratio computed from the nominal values must also have an associated range of uncertainty. Statistical analysis based on the assignment of a numerical uncertainty to each variable can be used to determine a confidence level for various degrees of uncertainty in the conversion ratio. This approach has not been applied to the Thorium HWO CR, so a true numerical confidence level cannot be assigned to the calculated conversion ratio. It is possible, however, to identify the significant factors influencing the conversion ratio and to evaluate their importance.

η (Neutrons Produced/Fissile Absorption)

The high conversion ratio potential of the thorium system lies mostly in η of U-233. Averaged over the reactor spectrum to 10 Mev, η of U-233 is about 2.26 compared to about 2.03 for U-235. Considering data uncertainties and difficulties in computing the thermal spectrum, a plus or minus 0.02 is not unreasonable for η of U-233. Since thorium absorbs about 70% of the excess neutrons during core life, this would lead to approximately 0.014 uncertainty in conversion ratio. A second problem concerned with η arises from the use of U-235 as a startup fuel. Conversion ratios of successive cycles on the approach to equilibrium follow the average η and therefore depend on the rate of transition from U-235 to U-233. Thus, the conversion ratio of cycle 2 depends on that of cycle 1, etc. If cycle 1 has a low conversion ratio for some reason, the makeup U-235 to cycle 2 will be higher, the average η in cycle 2 will be lower, and the U-236 buildup during cycle 2 will be greater. All these factors lead to the conclusion that a reduction in the initial conversion ratio (say on addition of neutron-absorbing structural material) may compound itself and have a greater effect on subsequent cycles. It is not clear what the effect would be on the true equilibrium cycle, however, since it is independent of method of approach. Calculations have not been made to evaluate the effect of compounding errors on conversion ratio, and a numerical estimate would be conjecture at this time.

Thorium Cross Section

In addition to the high η of U-233, the thorium system has the advantage of a fertile material thermal cross section nearly two and one-half times higher than that of U-238. This is an obvious advantage in competition for the excess neutrons and contributes to the high conversion ratio. Thorium's thermal cross section, however, is not well known in the range of 0.1 to 20 ev. Measurements are required to fill in this gap. Thorium's self-shielded resonance integral is known to about 10%. The best estimate of the total uncertainty in the thorium cross section is 5%. An error of 5% would be roughly equivalent to about 0.015 in conversion ratio.

Parasitic Absorption

The use of low-cross-section structural materials in the HWOCR reduces the effect of the uncertainty in these cross sections. The buildup of fission products and heavy isotopes is more significant and has a larger associated uncertainty. A typical core loses about 25% of its excess neutrons to parasitic absorption over life. A 10% variation in these losses would be equivalent to about 0.021 in conversion ratio.

Reactor Leakage

Leakage is small in the HWOCR even though core calculations account for increased leakage due to power shaping. The effect of an uncertainty in the computed leakage would be minor compared to other effects. A 10% error in leakage translates into about 0.004 in conversion ratio.

Lifetime Calculation Technique

Although the HWOCR has bi-directional refueling, calculations have been based on a simple one-dimensional approximation which was tested against a three-dimensional synthesis model. As described in section 2.3.1.3, this method was also checked against a more refined one-dimensional model closely resembling ORNL's TONG code. Any error in this process would show up primarily in an incorrect matching of feed enrichment and discharge burnup, which would be equivalent to computing the wrong average k_{eff} as a function of time. A 1.5% error in k_{eff} due to the model would change the conversion ratio by about 0.03.

Reactivity Control

All calculations have assumed a just-critical reactor. In reality, some excess reactivity must be carried for control of power shape and maneuvering. The vertical control system described in AI-CE Memo 6⁹ is worth about 0.003 Δk . Assuming that the available reactivity for control averages 50% in the core, a loss of 0.003 in conversion ratio would result.

Other Effects

The aforementioned items would be expected to play the largest roles in conversion ratio uncertainty, but other factors should be considered to some degree. Pa-233 losses are still small at present specific powers, and reasonable uncertainties in this cross section would not be a problem. Thorium fast-fission will be affected somewhat by the geometry, but will not be a strong factor. Nonuniform discharge burnups will also affect the conversion ratio, but no attempt to estimate these effects has been made.

Proper combination of these error estimates is difficult because they are interdependent. Considering the estimated errors to be standard deviations, the estimated uncertainty for the first cycle is ± 0.043 ; allowing for factors not considered, an uncertainty of about 0.05 is reasonable. The uncertainty in the conversion ratio for subsequent cycles may be slightly greater, perhaps ± 0.06 to 0.07.

4.1.4.4. Specific Power Limits

The specific power and associated pin heat rating are key parameters in establishing the economic performance of a design. As noted in section 2.4.3, the relatively low heat rating for the oxide design, which is limited primarily by fission gas release, penalizes the economic performance. It affects both fuel cycle and plant capital costs. For small changes from the design value of pin heat rating, the power cost changes about 0.005 mill/kwhr per percent change in kw/ft. However, higher heat ratings necessarily require more stringent thermal-hydraulic performance. The associated higher core pressure differentials and lower outlet temperatures decrease the net plant efficiency and thereby lead to an economic optimum heat rating. An increase of 25% to 12.5 kw/ft average in the hot pin would reduce the power cost of the oxide design by about 0.10 to 0.15 mill/kwhr, and this heat rating would probably be within 0.05 mill/kwhr of the economic optimum. Further increases in the carbide pin heat rating would not significantly affect power cost.

Since the optimum design of the nested-cylinder fuel element has the maximum heat flux on all fuel element surfaces, the maximum specific power is dependent on the pressure tube radial power

distribution and the allowable heat flux. Therefore, the specific power can be altered by a change in the power profile or in the allowable heat flux. The radial power peaking is the factor most likely to be altered by the fuel element test program. The economic effect of specific power changes for the nested-cylinder design is similar to that for the clustered-pin cores.

4. 1. 4. 5. Value of Fissile Material

The value of fissile material obviously plays a major role in determining the fuel cycle cost in any reactor. For thorium fuel cycles, which require fully enriched fuel and a relatively large fuel inventory, the fuel value is the prime determinant. For example, burnup and inventory charges, which are directly related to fissile material value, constitute 50 to 70% of the Thorium HWO CR fuel cycle cost. Thus, relatively small changes in such predominant factors can significantly alter the results.

The reference values for the Thorium HWO CR evaluation given in section 4. 1. 1 define the U-233 value at \$14/gm. An alternate value of \$18/gm with a penalty of \$12/gm U-236 was suggested by ORNL. Both of these values were considered in evaluating the fuel cycle costs of the reference designs. In addition, calculations were performed in which the dollar value of the core at the start of each batch was a constant. The value at the end of the batch life was equal to the initial value minus the U-235 makeup cost at the reference value of \$12/gm. Fuel cycle costs were also obtained with U-233 valued at zero and \$7/gm to indicate the effect of variation in value of the bred fuel. The 30-year levelized results for the nested-cylinder core based on re-cycle of fissile and fertile materials are as follows:

	<u>U-233 Value, \$/gm</u>				<u>Variable to maintain constant core value</u>
	<u>0</u>	<u>7</u>	<u>14</u>	<u>\$18-12/gm U-236</u>	
Fuel cycle cost					
Investor-owned	1.028	1.108	1.189	1.237	1.192
Publicly-owned	0.916	--	0.945	0.966	--

In general, the results show that the \$18/gm U-233 with a \$12/gm U-236 penalty produces the highest fuel cycle cost and that the \$14/gm U-233 leads to costs essentially identical to those obtained with the core value held constant at the start of each batch loading. The trends are similar for all three designs. For investor-owned conditions the fuel cycle costs may be expressed as a function of U-233 value by the equation

$$\text{fuel cycle cost in mills/kwhr} = 1.028 + 0.0115 (\$/\text{gm U-233}).$$

The value of startup and makeup fissile material (U-235 or plutonium) has a greater effect on fuel cycle cost than does the value of bred fuel alone, since the U-233 price is currently established by the U-235 value. Although costs of \$8/lb for U_3O_8 (yellow-cake) and \$30/kg for separative work have been used as the reference values, lower costs are anticipated in the near future which will reduce the value of fully enriched U-235 below \$12/gm. In addition, the value of plutonium may be partially determined by future technological developments and may differ from the reference value of \$10/gm. Calculations were made to assess the variation in Thorium HWOCR fuel cycle costs with decreasing startup fissile material values. Typical results are tabulated below. In obtaining these results for the nested-cylinder core, U-233 was held at 14/12 the value of U-235 for U-235 startup cases and at \$14/gm for plutonium startup cases. Comparable results for the oxide pin core with plutonium startup are shown in Figure 4-2.

	Startup fuel, \$/gm			
	U-235	U-235	Pu	Pu
	12	6	10	0
	Fuel cycle cost, mills/kwhr			
	U-235	U-235	Pu	Pu
Investor-owned	1.189	0.808	1.1818	0.5982
Publicly-owned	0.945	0.684	0.9408	0.4453

In general, the results for the investor-owned case indicate that fuel cycle costs decrease 0.064 mill/kwhr for each dollar per gram decrease in value of U-235. The results are defined by the equation:

$$\text{fuel cycle cost in mills/kwhr} = 0.427 + 0.0636 (\$/\text{gm U-235})$$

Under publicly-owned financing rates, the variation is 0.044 mill/kwhr per unit change in U-235 value. Thus, at U_3O_8 (yellowcake) and separative work values corresponding to \$6/gm U-235, one-half the reference value, fuel cycle costs for the thorium nested-cylinder design are 0.81 mill/kwhr at a 10% financing rate and 0.68 mill/kwhr at a 5% rate. Comparable values at one-half the reference value of plutonium are 0.890 and 0.693 mill/kwhr, as shown above. Although changes in value of this magnitude are unlikely to occur, some changes from the reference conditions are expected. These results indicate approximate changes in the Thorium HWOCR fuel cycle cost with modified fuel material values.

4.1.4.6. Film Fouling

For the purposes of the design evaluation, the maximum allowable prefouling surface temperature was set at 830 F for the clustered-pin designs and 820 F for the nested-cylinder design. (The nested-cylinder value is less because of the greater allowable heat flux.) A change in surface temperature will significantly modify the thermal performance, as indicated in Figure 4-3. The improved thermal characteristics, in turn, will increase the net plant efficiency, reduce the reactor size, and significantly improve the economic performance. A 50-F increase in surface temperature would modify the operating parameters extensively and reduce the power cost by about 0.05 to 0.10 mill/kwhr. By contract, B&W is dependent on AI-CE for coolant technology; AI-CE is conducting an experimental program to investigate film fouling and resolve this uncertainty.

4.1.5. Conclusions of Economic Studies

Thirty-year levelized fuel cycle costs are essentially identical for all three reference designs; under investor-owned financing conditions the variation is only 0.04 mill/kwhr. The fuel cycle costs are about 1.2 mills/kwhr for investor-owned financing and 0.95 mill/kwhr for publicly-owned financing for either U-235- or plutonium-fueled cores at reference materials values. The fuel-management method selected to check the validity of the fuel cycle cost, which are primarily based

on batch recycle nuclear data, confirmed the general accuracy of the batch recycle economics results.

The value of materials, the financing rate, and the technical limits all influence the fuel cycle cost. The more important technical limits are fuel exposure or burnup, specific power, and clad surface temperature; the economic effects of these limits have been evaluated. The fuel cycle costs decrease about 0.05 mill/kwhr for each percentage decrease in the financing rate. Fuel material values are major determinants of fuel cost. Under publicly-owned conditions with fuel material at one-half the reference value, fuel cycle costs as low as 0.68 mill/kwhr could be realized.

Plant capital, operation and maintenance, and D₂O costs constitute approximately two-thirds of the total power costs under investor-owned conditions; the percentage of these costs is slightly smaller under publicly-owned conditions. Capital and D₂O costs for the nested-cylinder design are about 0.15 mill/kwhr lower than these costs for the clustered-pin designs. This differential is the major factor in the favorable economic position of the nested-cylinder design. Power costs for the three reference designs are in the range of 3.5 to 3.7 mills/kwhr under investor financing and 2.4 to 2.5 mills/kwhr under public financing. Since the power cost differentials fall within the uncertainty band, other factors must be given careful consideration in selecting a fuel element design for future development.

4.2. Fuel Utilization

The term, fuel utilization, is normally considered a measure of the efficiency of the use of natural resources of fissionable material. In this respect, the Thorium HWO CR has excellent fuel utilization characteristics. In addition, there are strong incentives to exploit the vast energy resources latent in thorium. The Thorium HWO CR is the most practical vehicle for achieving wide-scale utilization of thorium.

For evaluating the fuel utilization of the HWO CR, AI-CE uses the Dynamic Utilization as the figure of merit. The following definitions and equation are taken from AI-CE MEMO-6.⁹ Assuming the nuclear power industry is growing at an exponential rate characterized by a doubling time, T_d , the defined quantities are:

1. The Exponential Investment (E.I.) of natural uranium is the ratio of the number of tons of natural uranium as U_3O_8 fed into the industry up to any time, t , for both burnup and inventory, to the number of megawatt years of electric energy produced up to the time, t . Obviously, the value of the E.I. depends upon the doubling time. An appropriate doubling time is necessary to determine an E.I. value that has a meaningful relationship to the fuel supply problem.

2. The Dynamic Utilization (D.U.) of natural uranium is the reciprocal of the E.I. The D.U. is proposed as the figure of merit, evaluated for a suitably selected doubling time.

Assuming $e^{0.693 t/T_d} \gg 1$, the E.I. can be calculated with the following equation:

$$E.I. = \frac{N}{W} = B + \frac{0.693 S}{FT_d}$$

where

S = specific inventory including outcore requirements = tons of U_3O_8 required to provide the U-235 for 1 MWe of installed reactor capacity

F = plant factor

B = specific makeup for burnup in equivalent tons of U_3O_8 per megawatt-year of electric energy produced

W = total electric energy produced from $t = 0$ to $t = T$

N = total natural uranium, as tons of U_3O_8 , put into the system for burnup plus inventory from $t = 0$ to $t = T$

Two trends may be observed from the foregoing equation:

1. The D.U. is sensitive to the assumed doubling time of the nuclear economy. Generally, a short doubling time favors reactors with low specific inventory, such as the Uranium HWO CR. Long doubling times are more favorable to a reactor with low specific burnup, such as the high-conversion-ratio Thorium HWO CR.

2. The D.U. considers the number of tons of U_3O_8 fed into the industry up to any time, t . It does not consider, however, the tons

committed for operation of the reactors beyond time t ; e. g. , all reactors going on the line at time t require an ore commitment for 30 years of operation. Therefore, the D. U. favors the low-inventory reactor rather than the good converter.

The D. U. does provide one basis for comparing the relative fuel utilization characteristics of the thorium-optimized 1000 MWe HWOCR's. Examination of Table 4-6 shows only slight differences in the D. U. for the three designs. The differences are within the accuracy of the calculations, so that no single design has a conclusive advantage in D. U. Similarly, the conversion ratios for all three reactors are essentially the same (0.86 to 0.87), which seems to indicate that designs optimized for economics have about the same conversion ratio. The uncertainties in the conversion ratio were discussed in section 4.1.

Available data on the "idealized" cycle conversion ratios of the Thorium HWOCR designs have been correlated against the important design variables. The following equation was found to predict the conversion ratio from the "idealized" cycle nuclear calculations for clustered-pin cores:

$$\text{C.R.} = 0.86 + 0.352 V_f - 2.6 \times 10^{-6} (\text{MWD/T}) - 1.4 \times 10^{-3} (\text{w/gm})$$

where

C.R. = conversion ratio

V_f = volume fraction of thorium metal in the pressure tube

The correlation of oxide and carbide conversion ratio data is shown in Figure 4-4.

The conversion ratio is less sensitive to the fuel volume fraction in the nested-cylinder design than it is in the clustered-pin cores. The variation of conversion ratio with volume fraction in the nested-cylinder designs is also shown in Figure 4-4. These results can be expressed by the following equation:

$$\text{C.R.} = 0.944 + 0.21 V_f - 2.6 \times 10^{-6} (\text{MWD/T}) - 1.4 \times 10^{-3} (\text{w/gm})$$

This difference in sensitivity is probably due to the different methods of changing volume fractions in the two designs. Fuel is added or removed uniformly over the pressure tube area by changing the pin pitch or fuel pin diameter in the clustered-pin elements. In contrast, the

nested-cylinder fuel volume fraction is changed by adding or removing fuel preferentially toward the center of the element to maintain a thermally optimum design. Consequently, the fuel fraction is changed in a relatively low flux region, near the center of the element, while the fuel fraction remains constant in the outer, higher flux region of the pressure tube. These results indicate that the effect of fuel volume fraction on conversion ratio is flux dependent and that for the same fuel volume fraction the removal of fuel to incorporate a center moderator is not as detrimental to conversion ratio as is the removal of fuel uniformly over the pressure tube area. Therefore, the optimization studies have generally shown that higher conversion ratios can be more readily achieved in a thorium metal nested-cylinder core than in the pin cores.

The correlations also indicate that the conversion ratio decreases with increasing burnup and specific power, but the decrease is slow enough to permit economic optimization without large penalties in the conversion ratio. It is significant that the conversion ratio is sufficiently sensitive to the volume fraction of thorium that a near-optimum design requires a relatively high-inventory, high-pressure-drop core for good economy and fuel utilization.

Another fuel utilization aspect of the Thorium HWOCR is the ability to use plutonium for startup and makeup. The availability of low-cost plutonium and the late development of the fast breeders would make the plutonium-thorium system especially attractive. The use of plutonium in place of uranium as fissionable material will probably not introduce significant problems, as discussed below and in greater detail in section 3.

In the metallic fuels, plutonium is in solid solution at the temperatures and composition under consideration for the HWOCR. The microstructure is monophasic alpha-thorium until uranium is bred into the alloy; upon achieving 1.5% uranium, the precipitation of gamma-uranium commences. This alloy should operate as the comparable thorium-uranium alloy operates.

In the oxide fuel, the operation of PuO_2 is expected to be similar to the operation of UO_2 , as evidenced by irradiation experiments. PuO_2 and UO_2 are isomorphous and otherwise chemically and physically similar.

The quaternary, (Th, Pu, U)O₂, developed by the breeding in of uranium would probably behave as the ternary behaves in the region of interest.

In the monocarbide fuel, although there is very little data on (Th, Pu)C, there is no reason to suspect aberrant behavior. ThC, PuC, and UC are isomorphous and similar in chemical and physical properties. The quaternary, (Pu, Th, U)C, in the concentrations of interest for the HWO CR would not be expected to deviate the behavior markedly from the (Th, U)C ternary.

On the basis of the preceding discussion, all three of the thorium fuel forms offer excellent fuel utilization potential, but no one form has a clear-cut superiority over the others.

4.3. Compatibility of Thorium Fuel Elements With AI-CE HWO CR

4.3.1. Relative Performance

As discussed in section 2, all three thorium fuel element types are basically compatible with the 1000 MWe Uranium HWO CR. However, varying economic penalties are incurred in adapting the elements to the HWO CR design, depending on the fuel material and the assumptions or methods used in making the adaptation. By definition, the governing thorium carbide properties are similar to those of uranium carbide, and the thorium monocarbide fuel element is specifically designed for the Uranium HWO CR.

The nested-cylinder design can be adapted to the 1000 MWe HWO CR without imposing an economic penalty on fuel cycle cost provided the full power capability of the element is utilized. The thorium metal nested-cylinder fuel element does have a capability for more power output per unit of volume than a uranium carbide fuel element does. Therefore, not all of the pressure tubes in the AI-CE HWO CR are needed to achieve the same power output with nested-cylinder fuel elements. If the secondary equipment were sized for a "stretch" capability, the use of thorium metal nested-cylinder fuel elements in the AI-CE HWO CR would allow an increase in the station power rating and would improve the economic performance.

The thorium oxide fuel element has a relatively low linear heat rate in comparison with the uranium carbide fuel element. The thorium oxide element can be adapted to the AI-CE HWO CR by using 55 pins instead of 37 pins, as noted in section 2.4. However, the smaller pin

size required is less than the optimum and imposes a fuel cycle cost penalty of about 0.10 mill per kwhr.

Thus, the thorium fuel materials and both pin and nested-cylinder geometries are basically compatible with the AI-CE HWO CR design. Additional effort to define the best method of achieving the necessary modifications could further reduce the already small economic penalties.

4.3.2. R&D Spinoff

Information on the selection, specification, procurement, fabrication, properties, characteristics, and irradiation behavior of SAP will necessarily be obtained from AI-CE if thorium dioxide or monocarbide is selected as the fuel material. Similarly, engineering tests performed by AI-CE will be generally applicable to the clustered-pin thorium fuel elements.

If thorium monocarbide is chosen, AI-CE's fabrication experience with uranium monocarbide will provide a useful reference basis for B&W's investigations. Specifically, knowledge of methods of casting, finishing, and inspecting pellets will be required.

AI-CE's experience with WR-1 irradiations will also prove beneficial to the B&W program. Applicable aspects are design and hazards evaluation of the experiment, fabrication of the bundle, and working arrangements with AECL/WNRE.

The R&D spinoff is reflected in the development cost estimates given in section 4.4.

4.4. Technical Feasibility

The test programs are discussed in section 3. To enable direct comparison, this section briefly summarizes the key technical problems and the probability of their solution, and the preliminary schedules and cost estimates for the development programs.

4.4.1. Key Technical Problems

4.4.1.1. Thorium Metal

As to the fabrication of fuel elements, the co-extrusion conditions for large-diameter, thin-walled cylinders have not been fully established. The process will require development to ensure dimensional reliability and security of the end-closure. The control of

coolant channel gaps with spacing devices poses another problem, which cannot be fully assessed until irradiation testing is performed. Proof testing is also required to determine the effect of irradiation on the dimensional stability of the coolant channel.

4.4.1.2. Thorium Dioxide

Further investigation is required to determine the thermal-conductivity and gas-release properties of the dioxide in both pellet and powdered forms. The thermal conductivity investigation is particularly troublesome because of the possibility of large errors in the determination, the high temperatures that must be achieved, and the added difficulty of working with powder. The buildup of gas pressure within the low-ductility SAP cladding could cause clad rupture and will require careful surveillance.

4.4.1.3. Thorium Monocarbide

The principal difficulty in fabricating thorium monocarbide is its reactivity with moisture. Therefore, thorium monocarbide must be stabilized by alloying, or dry-processing techniques must be used. After establishing the best route, casting, finishing, and inspection procedures must be developed. Since no neutron irradiation of thorium monocarbide has been reported, the dimensional and chemical stability and gas release properties must be established.

4.4.2. Probability of Successful Development

4.4.2.1. Thorium Metal

The fabrication unknowns probably can be resolved within the time allowed. However, the effect of irradiation on the coolant channel gaps will require extensive testing to determine whether a problem exists; if there is a problem, a long and costly procedure may be required to solve it. We feel that the technology to be developed in the proposed development program would help solve such a problem.

4.4.2.2. Thorium Dioxide

The thermal conductivity may not be conclusively determined in this program because the techniques and methods of interpretation are controversial. In any case, the program should

make a significant contribution to the overall solution and make available working values for reactor designers. The gas-release and pressure-buildup problem will be fully appreciated only after irradiation testing; any existing problems probably could be solved by introducing additional plenum space, using less gaseous feed material (may require development), or by using pellets instead of powder (introduces other difficulties). It is felt that current technology can provide the ultimate solution.

4. 4. 2. 3. Thorium Monocarbide

Owing to the low uranium content of the monocarbide, we may not be able to achieve stabilization without excessive alloying. However, this point can be readily determined, and dry-box procedures can be used if necessary. There should be no irradiation problems other than those arising from the use of uranium monocarbide. There is still a great deal of uncertainty associated with this material, and the large economic penalty imposed by dry-box procedures and the unknown effect of neutron irradiation reduce the chance of using monocarbide successfully.

4. 4. 3. Development Costs and Schedules

The fuel element test programs are defined in section 3. Preliminary cost estimates for these programs are summarized in Table 4-7, and schedules are shown in Figures 4-5, 4-6, and 4-7. All the test programs are scheduled for completion in FY-1972. Initial WR-1 irradiations of thorium-oxide and thorium-metal (multicylinder) fuel elements could begin in 18 months, considering USAEC FY-1967 budgeting restraints. Thorium monocarbide irradiation could begin in two years. The total cost estimates for the fuel element test programs through FY-1972 are \$5,005,300 for thorium oxide, \$7,366,300 for thorium metal, and \$7,123,800 for thorium monocarbide. These estimates are preliminary and are primarily intended to show the relative costs of developing these fuel elements as a factor in the selection of a reference design.

As discussed in section 3.5, some R&D (not included in the fuel element test programs) in the areas of experimental physics and recycle technology will be required to realize commercial acceptance of the HWOCR thorium fuel cycle.

The total experimental physics program outlined in section 3.5 is estimated to cost approximately \$1.8 million and to require three to five years. There is no appreciable difference in cost among the three fuel element types being considered. It is assumed that the physics work will be done at the Savannah River Laboratory.

Relative costs only were estimated for recycle technology development. These estimates are based on two major assumptions:

1. The TRIP facility process flow sheets to be developed are those reported in BAW-393-3⁶ for metal and oxide; the carbide flow sheet is similar for reprocessing, but uses essentially pellet technology in canning cast monocarbide slugs.

2. Except for the problem areas listed in section 3.5, the current state-of-design technology is adequate to allow the TRIP facility to be designed today with a reasonable probability of success.

Based on these assumptions, the relative development costs were evaluated for each major phase of fuel recycle and are tabulated below.

<u>Fuel type</u>	<u>Relative recycle development cost</u>		
	<u>Reprocessing</u>	<u>Conversion</u>	<u>Fabrication</u>
Oxide	X	Y	Z
Metal	X	2Y	5Z
Carbide	2X	5Y	3Z

To arrive at final recycle development costs, a weighting was assigned to the separate steps of recycle. This weighting was 0.35 for reprocessing, 0.15 for conversion, and 0.50 for fabrication. Applying this weighting to the values just given, the final relative development costs are:

Oxide	X (dollars)
Metal	3X (dollars)
Carbide	3X (dollars)

4.5. Safety

The stability characteristics of the HWO CR should be similar to those of other large D₂O-moderated reactors except that the use of neutron-absorbing organic coolant leads to positive void coefficients. It is assumed that the control system designed by AI-CE for the Uranium HWO CR will be applicable to the thorium system. Thorium, because of its higher cross section, tends to provide a more stable system; conversely, the poison control material has less reactivity effect in controlling an oscillation in the thorium system than in the uranium system. Preliminary two-dimensional control studies indicate that the Thorium HWO CR will respond well to a control system similar to that proposed by AI-CE. The question of Pa-233 decay to U-233 during a shutdown and the corresponding additional load placed on the control system is recognized, but has not been evaluated yet. (A 30-day shutdown would decay about half the protactinium and inject approximately 1.5% $\Delta k/k$ into a typical core.)

A basic question arises as to whether any one of the three thorium fuel element designs is better suited to the HWO CR because of safety or control considerations. At this stage, no grounds have been established that would allow us to assume that any reactor dynamic characteristic will be sufficiently magnified or suppressed to cause the selection or rejection of a design on the basis of control or safety. This is not to say that all three fuel element types have the same coefficients or transient response, but that there are not enough differences to influence the choice of a design for further development. Conductivity differences among the three fuel matrices will yield varying Doppler coefficients. The nested-cylinder design was not expected to have a Doppler coefficient comparable to those of the clustered-pin designs because of the high metal conductivity; however, ORNL calculations showed that the Doppler coefficient of the metallic cylinder is more effective than that of the others.

Relative isolation of the coolant channels in the nested-cylinder element will undoubtedly produce void reactivity characteristics that are different from those of the open-lattice clustered-pin element with associated interchannel coolant mixing. Contributions external to the pressure tube, however, will affect all fuels in much the same manner, that is, they will affect D₂O purity, D₂O voiding, D₂O temperature,

leakage, control material insertion, etc. Only a detailed study of a given design will ascertain the dynamic characteristics and the suitability of a fuel element for HWOCR application. Such information is not now available for any design and, therefore, cannot be used as a factor in the evaluation. Safety studies of this type are planned for FY-1967.

4.6. Commercial Acceptance

The prime consideration of the electric utility industry in the United States in evaluating fuels is the power generation cost. In this respect, based on a reasonable extrapolation of existing technology, thorium metal fuel elements in a 1000-MWe HWOCR will provide power cost savings of about 0.15 mill/kwhr over oxide or carbide. This savings, if realized in commercial practice, is clearly sufficient to justify utility preference for thorium metal.

However, the United States' nuclear economy is geared to the UO_2 fuel cycle. To reach a similar commercial position, wide-scale acceptance of thorium fuels by the electric utility industry is necessary. Such acceptance will require proven performance of the fuel cycle. In this respect, the transferability of UO_2 experience makes ThO_2 the most likely thorium candidate to win commercial acceptance in the United States market.

In foreign countries having no enrichment facilities, the prime objective of a thorium reactor would be to achieve minimum fissile material requirements. Therefore, the goal is a thorium breeder or near-breeder. As discussed in section 4.2, the conversion ratio is strongly dependent on the fuel volume fraction within the pressure tube. The designs optimized for economics have about the same conversion ratio; however, optimization studies indicate that near-breeder performance can be most readily achieved with the thorium metal nested-cylinder design. Commercial application of a thorium metal fuel cycle, including recycle, in a foreign country would face difficulties similar to those expected in the United States.

4.7. Summary

The three thorium fuel element types have been evaluated according to the factors that are pertinent to the objectives of the HWOCR Thorium Fuel Cycle Development Program. The lack of data in many

areas of thorium technology makes it difficult to reach sound conclusions. Therefore, the uncertainties in the results have also been evaluated. The conclusions reached for individual evaluation factors provide the bases for weighing alternate courses of action as reported in section 5.

Table 4-1. 30-Year Levelized Fuel Cycle Cost,
U-235 Startup Fuel

	Cost, mills/kwhr					
	Thorium metal nested-cylinder core		Thorium oxide pin core		Thorium carbide pin core	
	Investor- owned	Publicly- owned	Investor- owned	Publicly- owned	Investor- owned	Publicly- owned
Fabrication expense	0.2536	0.2528	0.2570	0.2542	0.3079	0.3076
U-235 expense	0.4525	0.4220	0.4548	0.4199	0.4431	0.4128
U-233 expense	-0.1694	-0.1404	-0.2047	-0.1718	-0.1711	-0.1424
Reprocessing & shipping expense	0.1668	0.1668	0.1731	0.1731	0.1757	0.1756
Fabrication working capital charge	0.0332	0.0166	0.0412	0.0203	0.0419	0.0209
U-235 working capital charge	0.1477	0.0709	0.1675	0.0797	0.1396	0.0669
U-233 working capital charge	0.3300	0.1693	0.3681	0.1898	0.3248	0.1664
Reprocessing & shipping working capital charge	-0.0260	-0.0131	-0.0314	-0.0159	-0.0279	-0.0141
Total	1.1885	0.9449	1.2256	0.9493	1.2339	0.9937

Table 4-2. 30-Year Levelized Fuel Cycle Cost,
Plutonium Startup and Recycle

	Cost, mills/kwhr			
	Oxide pin core		Nested-cylinder core	
	Investor- owned	Publicly- owned	Investor- owned	Publicly- owned
Fabrication expense	0.2460	0.2464	0.2456	0.2457
U-235 expense	0.4819	0.4520	0.4706	0.4434
U-233 expense	-0.2166	-0.1854	-0.1888	-0.1611
Reprocessing & shipping expense	0.1706	0.1706	0.1661	0.1662
Fabrication working capital charge	0.0394	0.0196	0.0324	0.0162
U-235 working capital charge	0.1527	0.0720	0.1338	0.0636
U-233 working capital charge	0.3796	0.1974	0.3479	0.1799
Reprocessing & shipping working capital charge	-0.0307	-0.0155	-0.0259	-0.0131
Total	1.229	0.9572	1.1818	0.9408

Table 4-3 Capital, D₂O, and Operation and Maintenance Cost —
1000-MWe Thorium-Optimized Designs

	No. pressure tubes	Reactor power	Net efficiency	Cost, \$ × 10 ⁻⁶					Cost, mills/kwhr			
				Reactor hardware	Pump and driver	Nuclear instruments	Total capital	D ₂ O	D ₂ O	Operating	Capital	Total
Th metal — investor	324	3137	33.76	6.65	4.28	1.13	119.14	9.87	0.155	0.3	1.900	2.355
Th metal — public	324	3137	33.76	6.65	4.28	1.13	115.28	9.87	0.090	0.3	1.070	1.460
ThC — investor	548	3152	34.14	9.34	3.67	1.90	123.13	14.20	0.216	0.3	1.959	2.475
ThC — public	548	3152	34.14	9.34	3.67	1.90	119.14	14.20	0.122	0.3	1.106	1.528
ThO ₂ — investor	758	3100	34.70	11.90	2.70	2.63	126.37	12.66	0.194	0.3	2.011	2.505
ThO ₂ — public	758	3100	34.70	11.90	2.70	2.63	122.28	12.66	0.110	0.3	1.135	1.545

Table 4-4. Batch "Idealized" Cycle Fuel Cycle Costs

	<u>Nested cylinder</u>			<u>Carbide pin</u>	<u>Oxide pin</u>	
100-MWe Th-optimized design, MWD/T	11,550	14,920	19,300	22,000	--	22,450
<u>Fuel cycle cost, mills/kwhr</u>						
Investor-owned	1.199	1.121	1.067	1.083	--	1.070
Publicly-owned	0.962	0.899	0.853	0.879	--	0.832
Thorium core for AI-CE 1000-MWe HWOCR, MWD/T	17,500	20,300	--	22,000	16,360	19,300
<u>Fuel cycle cost, mills/kwhr</u>						
Investor-owned	1.102	1.075	--	1.083	1.265	1.200
Publicly-owned	0.881	0.857	--	0.879	1.089	1.031

Table 4-5. Comparative Unit Processing Costs for Thorium-Optimized HWO CR Fuel Elements

Fuel type	Total unit processing cost, \$/kg ^(a)	
	ORNL estimates compiled at B&W	B&W estimates
Alloy nested-cylinder ^(b)	67	73
Oxide pin ^(c)	74	70
Carbide pin ^(d)	81	80

(a) Includes reprocessing, conversion, fabrication, and shipping.

(b) Four-cylinder, center moderator, 5-inch-ID pressure tube.

(c) 37-pin, 0.510-inch ID, 4.03-inch pressure tube.

(d) 37-pin, 0.493-inch ID, 4.03-inch pressure tube.

Table 4-6. Comparison of Dynamic Fuel Utilization

Utilization factor	Thorium HWO CR		
	4-cylinder Th metal	37-pin Th oxide	37-pin Th carbide
Specific inventory, tons U ₃ O ₈ /MWe	0.74	0.85	0.71
Specific burnup, tons U ₃ O ₈ /MWe-yr	0.052	0.047	0.054
Dynamic fuel utilization, MWe-yr/ton U ₃ O ₈ , for nuclear economy doubling time:			
5 yr	5.5	5.2	5.6
7 yr	7.0	6.0	7.0
10 yr	8.5	8.3	8.7

Table 4-7. Fuel Element Test Program Preliminary Cost Estimate Summary (\$)

1. Thorium Metal Nested-Cylinder Fuel Element

Task	Fiscal year						Total
	67	68	69	70	71	72	
21 - Preliminary Design	150,000	110,000	110,000	60,000	--	--	430,000
22 - Fabrication Development	257,100	635,600	625,800	422,800	177,800	--	2,119,100
23 - Fuel Characteristics	--	431,100	537,100	476,900	357,600	76,600	1,879,300
24 - Engineering Tests	--	175,000	229,000	224,000	--	--	628,000
25 - Irradiation Test	74,600	274,100	476,500	276,000	523,100	65,600	1,689,900
26 - Final Design	--	--	--	175,000	300,000	145,000	620,000
Total	481,700	1,625,800	1,978,400	1,634,700	1,358,500	287,200	7,366,300

2. Thorium Dioxide Clustered-Pin Fuel Element

Task	Fiscal year						Total
	67	68	69	70	71	72	
21 - Preliminary Design	150,000	110,000	110,000	60,000	--	--	430,000
22 - Fabrication Development	220,900	85,900	35,000	--	--	--	341,800
23 - Fuel Characteristics	--	467,800	763,700	403,100	62,700	--	1,697,300
24 - Engineering Tests	--	--	42,000	124,000	65,000	--	216,000
25 - Irradiation Test	82,100	154,300	361,100	321,800	467,300	298,600	1,685,200
26 - Final Design	--	--	--	175,000	300,000	145,000	620,000
Total	453,000	818,000	1,311,800	1,083,900	895,000	443,600	5,005,300

3. Thorium Monocarbide Clustered-Pin Fuel Element

Task	Fiscal year						Total
	67	68	69	70	71	72	
21 - Preliminary Design	150,000	110,000	110,000	60,000	--	--	430,000
22 - Fabrication Development	207,800	667,200	625,000	650,000	--	--	2,150,000
23 - Fuel Characteristics	--	387,700	529,600	292,200	194,100	88,100	1,491,700
24 - Engineering Tests	--	--	42,000	124,000	50,000	--	216,000
25 - Irradiation Test	131,300	333,700	356,400	505,900	589,800	299,000	2,216,100
26 - Final Design	--	--	--	175,000	300,000	145,000	620,000
Total	489,100	1,498,600	1,663,000	1,807,100	1,133,900	532,100	7,123,800

Figure 4-1. Fuel Cycle Cost Vs Fuel Exposure

Figure 4-2. Fuel Cycle Cost Vs Plutonium Value (Oxide Core)

Figure 4-3. Effect of Allowable Cladding Surface Temperature on Core Pressure Drop and Coolant Outlet Temperature

Figure 4-4. Thorium Fuel Elements: Conversion Ratio Factor Vs Fuel Volume Fraction

Figure 4-5. Preliminary Schedule for Thorium Metal
Nested-Cylinder Fuel Element Test
Program

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Figure 4-6. Preliminary Schedule for Thorium Oxide Clustered-Pin Fuel Element Test Program

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Figure 4-7. Preliminary Schedule for Thorium Monocarbide Clustered-Pin Fuel Element Test Program

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(Section 4)

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5. RECOMMENDATION

The advantages and disadvantages of developing each type of fuel element are summarized in Table 5-1. A review of these factors leads to a recommendation.

The most obvious conclusion is that there is little incentive for developing thorium monocarbide. The lack of thorium monocarbide data required the assumption of uranium monocarbide characteristics in many areas. On this basis, the thorium monocarbide has about the same economic and fuel utilization performance as thorium oxide. In addition, the thorium monocarbide is at least an order of magnitude more reactive in air than uranium monocarbide. Therefore, ThC must be handled in a dry, inert atmosphere, or it must be alloyed with a stabilizing agent. Considering the potential development problems and the probable lack of economic or fuel utilization advantages, there is no justification for recommending the development of the thorium monocarbide fuel element.

On the basis of our studies, the thorium metal nested-cylinder fuel element offers the best potential for low cost power and efficient fuel utilization. The economic potential is based on the predicted thermal performance, which allows more power output per unit volume and a consequent improvement of about 0.15 mill/kwhr in capital and D₂O costs, in comparison with the thorium oxide core. The principal questions in the predicted performance are whether or not the large thin-wall, multicylinder fuel elements can be built economically to the required tolerances and can maintain dimensional stability under long-term (20,000 MWD/T average) exposure in an HWO CR environment. These questions cannot be resolved without a development program.

As stated, the economically optimum thorium oxide HWO CR has predicted capital and D₂O costs about 0.15 mill/kwhr higher than those for the thorium metal HWO CR. However, the fuel cycle costs and the conversion ratios of the two designs are essentially equal. The pin heat rating (kw/ft), which is primarily responsible for the capital cost

differential, is limited by fission gas pressure buildup in the low-ductility SAP cladding combined with a relatively low external pressure. Resolution of the fission gas pressure problem could lead to higher pin heat ratings and longer fuel exposures, both of which would improve the economics of the thorium oxide HWOOCR.

In general, the technology of thorium fuels is inadequate to draw firm conclusions. Our recommendation must be based on analytical studies using existing data. From a technical standpoint, the logical development program will carry both thorium oxide and thorium metal fuel elements through several phases of irradiation testing aimed at determining the dimensional stability of the thorium metal cylinders and the fission gas pressure limits of the thorium oxide pins. With this specific HWOOCR-oriented test data available, the correct decision can be made with a high degree of confidence. The two-pronged test program would be a compromise between the separate programs outlined in section 3 and would emphasize obtaining the necessary irradiation performance data in a minimum time.

If practical considerations require the selection of a single fuel element for development, our recommendation is thorium oxide. The power cost difference between the oxide and the metal cases is within the uncertainty band, and the oxide costs can be predicted with more confidence than the metal costs can. The conditions under which thorium will be used in the HWOOCR are also uncertain. The development of a thorium oxide fuel element offers the shortest and cheapest route and the best probability of success for putting the United States in a position to use thorium in the HWOOCR when conditions are favorable. The thorium oxide fuel will require the least initial investment during the buildup of a thorium economy, and the fabrication process has the most flexibility to meet changing conditions (e.g., changes in the value of plutonium that may favor the Thorium HWOOCR). In conclusion, the thorium oxide fuel element has the best chance of gaining large-scale application at a minimum development cost.

Table 5-1. Evaluation of Alternate Courses of Action

Develop Thorium Metal Nested-Cylinder Fuel Element	
Advantages	Disadvantages
<ol style="list-style-type: none"> 1. Best economic potential of thorium HWOGR's. Estimated total power cost of 3.55 mills/kwhr is 0.15 mill/kwhr better than cost of oxide or carbide. 2. Best thermal performance capability. Has more power output per unit volume than clustered-pin fuel element. Therefore, provides a "stretch" capability for AI-CE HWOGR with thorium fuel elements. 3. Good potential for near breeder. High fuel density and adaptability for high fuel volume fraction make high conversion ratio attainable. 4. Different geometry and cladding from AI-CE HWOGR fuel element. Provides a backup fuel element that is not dependent on success of SAP development. 5. Annular fuel elements are being considered as advanced designs for AI-CE HWOGR for desalting application. Th metal fuel element development would be beneficial to development of annular uranium fuel element. 	<ol style="list-style-type: none"> 1. More design uncertainties than for oxide fuel element. Principal uncertainties are fabrication tolerances and irradiation stability of large thin-wall, multicylinder elements. 2. Higher development cost than for oxide. Fuel element test program cost is about \$3 million higher. Development cost of recycle technology is higher by a factor of about 3. 3. Commercial acceptability will be hampered by lack of existing economy for metallic fuels and annular geometry. 4. Changes in pressure tube power distribution with fuel burnup will require downgrading of optimum design since each cylinder is designed for maximum heat flux. Power distribution change is not significant with U-235 startup, but is with Pu startup. Cylinders are more difficult to zone load than pins. 5. Full potential of cylinders cannot be realized in AI-CE HWOGR unless some pressure tubes are blocked off or unless a stretch capability is built into the secondary system equipment. 6. Optimum cylinder design has 5-inch-ID pressure tube. Additional development and testing would be required for this size pressure tube since AI-CE design has 4.03-inch ID.
Develop Thorium Oxide Clustered-Pin Fuel Element	
Advantages	Disadvantages
<ol style="list-style-type: none"> 1. Most advanced stage of development, which means <ol style="list-style-type: none"> a. Best probability of success. b. Less time to demonstrate compatibility of thorium and uranium systems. c. More certainty in cost estimates. 2. Most likely route for commercial acceptance of thorium fuel cycle because of <ol style="list-style-type: none"> a. Least initial investment during buildup of thorium economy. b. Transferability of experience from UO₂ economy. 3. Good transferability of technology between following programs: <ol style="list-style-type: none"> a. USAEC Thorium Utilization Program. b. Seed and Blanket (Thorium Breeder) Reactor Program. c. Canadian Heavy Water Program (ThO₂ is only form compatible with H₂O or D₂O coolant). d. SAP Development and Engineering Tests from AI-CE Program. 4. Flexibility in fabrication process as follows: <ol style="list-style-type: none"> a. Adaptable to Pu startup fuel. b. Adaptable to enrichment variations and zone loading. 	<ol style="list-style-type: none"> 1. Low pin heat rating (kw/ft) which means <ol style="list-style-type: none"> a. Higher capital costs than for Th metal core. b. 55-pin fuel element in AI-CE HWOGR. Smaller diameter pins impose fabrication cost penalty (approximately 0.1 mill/kwhr). 2. Uncertainty of fission gas pressure buildup (in SAP-clad pins with low external pressure) limits performance. 3. Achieving near-breeder performance is more difficult with ThO₂ than with higher density Th metal.
Develop Thorium Monocarbide Clustered-Pin Fuel Element	
Advantages	Disadvantages
<ol style="list-style-type: none"> 1. Maximum spinoff from AI-CE Program as follows: <ol style="list-style-type: none"> a. Same geometry. b. Same cladding. c. Same basic manufacturing process (note disadvantage 2). 2. All R&D will contribute to basic ThC technology. 	<ol style="list-style-type: none"> 1. Not enough ThC data are available for a conclusive evaluation. However, work done showed no economic or fuel utilization incentives for ThC development. 2. ThC is highly reactive with moisture in air. Even with 1% zirconium added to (Th,U)C the weight gain in air is about eight times that of uranium monocarbide. 3. High development cost.