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HEAVY-WATER-MODERATED POWER REACTORS
COOLED BY LIQUID D₂O OR H₂O

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1. INTRODUCTION

1.1 Advantages of D₂O Reactors

This presentation is concerned with power reactors that are moderated by heavy water and cooled by liquid D₂O or H₂O. In these reactors, the coolant is confined in pressure tubes, and complete physical separation of the coolant from the moderator is maintained.

Heavy water reactors combine excellent neutron economy with highly rated fuel assemblies that can be made by low-cost fabrication techniques. As a result, they offer uniquely low fuel cycle costs and consume less mined fissionable material per unit of heat produced than does any other type of converter reactor. Natural uranium fuel cycles can be employed, and D₂O reactors have the best potential for operating on a self-sustaining or breeder cycle with thorium fuel.

Of the various possible coolants for D₂O-moderated reactors, liquid D₂O has received the most attention to date, partly because of the extensive experience that has been gained with this coolant. Liquid D₂O offers low neutron absorption and good heat transfer properties. Since relatively large amounts of D₂O can be tolerated in fuel assemblies without significant reactivity penalty, much flexibility in fuel design is afforded. The main disadvantage is the high cost of D₂O; the charges on D₂O inventory and the cost of D₂O losses are important items. Because of these costs liquid H₂O is of interest as an alternative coolant. Organic liquids can also be substituted for liquid D₂O.

1.2 Du Pont Experience

Du Pont experience with D₂O reactors dates back to 1950, when the company undertook for the AEC the design, construction, and operation of five D₂O-cooled-and-moderated production reactors at Savannah River. As part of the same contract Du Pont has been engaged since 1957 in the development of D₂O power reactors. The emphasis in this still-active program has been on D₂O cooling. Particular attention has been paid to fuel development and to the physics of D₂O reactor lattices. Substantial engineering development work has also been performed. The Heavy Water Components Test Reactor (HWCTR), a fuel test facility designed and built by Du Pont, was

placed in operation at Savannah River in 1962, and is currently employed for irradiation testing of fuel elements for D_2O -cooled reactors. The Du Pont program is funded by the AEC, and all of the unclassified technology developed in the program is available to the public.

1.3 Other D_2O Reactors

At present, five D_2O -cooled-and-moderated reactors are in operation at pressures and temperatures of direct interest for economic power generation. These are the CVTR at Parr, S. C.; the PRTR at Hanford; the NPD at Rolphton, Ontario; the HBWR at Halden, Norway; and the HWCTR. In addition, the 200-MWe CANDU is in advanced construction in Canada. All of these reactors are sources of additional technical data and operating experience.

1.4 Long-Range Role of D_2O Reactors

We believe that D_2O reactors are well qualified for a position of importance in the development of nuclear power in the U. S. because they offer the prospect of both competitive power costs and good fuel utilization. Table 1-I summarizes the present development status and a logical course for the future. Present emphasis is on achieving low-cost electric power. By the early 1970's, it appears possible to carry the concept through prototype demonstration and to place in operation a full-scale competitive reactor. Concurrent development work can be concentrated on cost reduction, improved fuel utilization, and D_2O -thorium breeder technology. The adaptation of D_2O reactors to thorium fuel will make it possible to convert to breeding when the pressure on uranium prices makes such a conversion advantageous. The emphasis in the 1980's would be on advanced D_2O converters with improved uranium utilization, and in the 1990's on economical D_2O -thorium breeders.

TABLE 1-I

D₂O Reactor Program

OBJECTIVE	THORIUM	URANIUM	FUEL TESTS (HWCTR)
1960		Develop Concept & Optimize	
Development		Develop Prototype	Uranium
1965	Develop Breeding Technology & 1000-MWe Reference Design	Construct Prototype (300 MWe)	Thorium
1970	Test in Prototype	Construct 1000 MWe	
Cheap Fuel & Power		Demonstrate	
1980		Power & Process Heat	
Good Uranium Utilization			
1990	Power & Process Heat		
Thorium Breeding			
2000

2. REACTOR DESIGN

2. Design Parameters for 1000-MWe Plant

Table 2-I summarizes the principal design parameters for a 1000-MWe D₂O-cooled reactor.^(2.1) These data apply for a reactor that is optimized for present-day construction costs and uranium prices. The reactor is characterized by a uniquely high thermal rating per fuel position and by a high degree of radial flux flattening to improve the uniformity of power distribution and to reduce the reactor inventory of fuel. The high power density stems from the use of nested-tube fuel assemblies which are being developed at Savannah River. The combination of the tubular fuel shape with high D₂O velocity makes it possible to achieve high specific powers and heat fluxes without exceeding safe limits on fuel temperature or heat transfer. Fuel ratings approaching those shown in Table 2-I have been piloted in irradiation tests, and fuel elements have operated repeatedly at the listed heat flux.

The fuel assembly consists of three tubes of uranium oxide clad with Zircaloy-2, which are spaced by longitudinal ribs that are welded onto the inner fuel sheaths. The fuel assembly is housed in a zirconium alloy pressure tube, which in turn is insulated from the cold moderator by a dead gas space. The fuel assembly dimensions are listed in Table 2-II (a cross-sectional drawing of the fuel is shown in Figure 5-1). The dimensions of the fuel tubes are chosen to balance the temperature rises and pressure drops along the subchannels and to make the burnout safety factors of the three tubes uniform.

Figure 2-1 shows the lattice arrangement in which 516 fuel assemblies and 37 control positions are arranged on a 10-inch-square pitch. The reactor core is surrounded by a 20-inch radial D₂O reflector and a 24-inch axial D₂O reflector.

A simplified flowsheet for the primary coolant system is shown in Figure 2-2. D₂O coolant at 1700 psi is heated from 264 to 300°C in the reactor. From the reactor it passes to a steam generator, which produces saturated steam at 680 psia. A detailed list of the reactor design parameters is given in Table 2-III.

For more advanced designs of D₂O-cooled reactors, it should be possible to improve the power density shown in Table 2-I by at least 25%. Corresponding reduction can then be achieved in reactor size, cost, and fuel inventory.

2.2 Optimization of Reactor Core Parameters

A computer code^(2.2) has been used to determine the combination of the design parameters that produces the minimum energy cost. This code has been developed for D₂O reactors cooled with liquid D₂O, boiling D₂O, and organic coolants at 300, 500,^(2.3) and 1000 MWe.^(2.1) A typical set of results is illustrated in Figures 2-3 and 2-4 for a 300-MWe D₂O-cooled reactor plant.^(2.4)

2.3 Economic Studies of Other Coolants

In order to assess the economic potential afforded by the use of coolants other than liquid D₂O, conceptual designs were developed for D₂O-moderated reactors cooled with boiling D₂O, organic liquids, liquid H₂O, and H₂O fog. The results of these studies and the estimates of the energy costs from these various reactor concepts are given in Ref. 2.7 for 300- and 500-MWe systems. Subsequent studies were made comparing liquid D₂O, boiling H₂O, and organic coolants for 1000-MWe plants.^(2.8) These studies show that only the organic coolant offers a significant economic advantage over the liquid D₂O system and that a savings of 1/2 mill/kwh may be achieved with the organic-cooled system. However, we do not believe that this 1/2-mill/kwh difference is sufficient to warrant the three- to four-year delay in the heavy water program that development of the organic system would entail.

2.4 Reactors for Process Heat Applications

Studies^(2.10,2.11) have been made of the use of heavy water reactors to produce large amounts of low-cost process heat such as might be required for distillation of sea water. These studies were carried out in conjunction with an inter-agency study^(2.12) on desalination plants. Included in our studies were (1) reactor designs with minimum capital and fuel costs to produce low-pressure steam and (2) reactor designs to produce electricity in a back-pressure turbine and use the by-product steam for process heat. In both cases the process steam is produced at 250°F, a suitable temperature for use in a desalination plant.

A dual-purpose plant with an 1100-MWt reactor can deliver 170 MW of electricity and 57×10^9 Btu/day of steam at 250°F. This amount of steam is suitable for a desalination plant in the range of 50 million gallons per day.

TABLE 2-I

Summary of Design Parameters for 1000-MWe Reactor

Thermal power	3500 MW
Number of fuel positions	516
Over-all average power per assembly	6.9 MW
Power of average flat zone assembly	7.9 MW
Maximum assembly power	10.0 MW
Fuel material	UO ₂
U ²³⁵ enrichment	1.2%
Average specific power, MW/tonne U	54
Maximum $\int kd\theta$ ^(a)	43 watts/cm
Maximum D ₂ O velocity	46 ft/sec
Maximum heat flux	(900,000 Btu/(hr)(ft ²)
Maximum cladding surface temperature	330°C
System pressure	1700 psi
Coolant inlet temperature	264°C
Coolant outlet temperature	300°C
Throttle steam pressure and temperature	650 psia/257°C
Net electrical power (at 1.5" Hg)	1010 MW

(a) Fuel rating is 43 w/cm during initial exposure. The fuel rating averaged over the life of the hottest fuel assembly is 37 w/cm.

TABLE 2-II

Fuel Assembly Dimensions for
D₂O-Cooled-and-Moderated Reactor

Calandria tube

OD	4.84 inches
ID	4.78

Pressure tube

OD	4.66
ID	4.22

Housing tube

OD	4.10
ID	4.04

Fuel tubes

Clad diameters

No. 1	OD	3.68
	ID	3.16
No. 2	OD	2.56
	ID	1.92
No. 3	OD	1.28
	ID	0.61

Active core length	15 ft
--------------------	-------

Average cladding thickness	0.025 inch
----------------------------	------------

TABLE 2-III

Design Parameters for 1000-MWe ReactorFuel Design

System pressure	1700 psi
Design pressure	2000 psi
Design temperature	320°C
Coolant inlet temperature	264°C
Average outlet temperature	300°C
Moderator temperature	60°C
Weight of UO ₂ fuel	22 lb/ft
UO ₂ density	92% of theoretical
Initial enrichment	1.2% U ²³⁵
Power of average flat zone assembly	7.9 MWt
Maximum assembly power	10.0 MWt
Average coolant flow to assembly at 264°C	705 gpm
Maximum coolant velocity	46 ft/sec
Maximum surface temperature	330°C
Maximum $\int kd\theta$, fuel tube no. 1	32 watts/cm
Maximum $\int kd\theta$, fuel tubes no. 2 and 3	43 watts/cm
Maximum heat flux	900,000 Btu/(hr)(ft ²)

Core Design

Thermal power	3500 MW
Moderator power	142 MW
Number of fuel positions	516
Number of coolant loops	6
Coolant piping material	Carbon steel
Total flow at 264°C	350,000 gpm
Fuel inventory	68 tonnes U
Average specific power	54 MW/tonne U
Average fuel exposure	15,000 MWD/tonne U
D ₂ O inventory	680 tons

Steam Generator Design

Heat transfer area	350,000 ft ²
Steam pressure	680 psia
Steam temperature	260°C
Input power, including pump heat	3547 MW

Turbine Design

Throttle pressure	650 psia
Gross efficiency	31%
Gross electrical power	1100 MW
Power used by auxiliaries	90 MW
Net electrical power	1010 MW

Tank Diameter 26 feet

Tank Height 19 feet

○ Fuel Position (516)

● Control Position (37)
10-inch Square Pitch

FIG. 2-1 LATTICE FOR 1000-MWe D₂O-MODERATED REACTOR

FIG. 2-2 PRIMARY COOLANT FLOWSHEET FOR 1000-MWe D_2O -MODERATED REACTOR

FIG. 2-3 VARIATION OF ENERGY COSTS WITH COOLANT TEMPERATURES IN D_2O -COOLED REACTOR

FIG. 2-4 EFFECTS OF POWER FLATTENING AND REFLECTOR THICKNESS ON ENERGY COSTS FROM D_2O -COOLED REACTOR

3. SIZE CONSIDERATIONS

3.1 Full-Scale Reactors

Heavy water reactors have relatively large cores and high power capability per individual fuel position. These reactors are most economical when they are made large enough to overcome the disadvantage of the large cores and to exploit fully the advantages of the high power capability. Present indications are that D₂O-cooled reactors will be only marginally competitive at capacities below about 500 MWe, and will find their principal application at or above this size level.^(3.1) Smaller reactors are technically feasible, but unit capital cost increases so rapidly with decreasing size that there is little incentive to consider reactors of low capacity except possibly when financing charges are exceptionally favorable. Figure 3-1 shows the cost savings that are realized from increases in plant sizes from 300 to 1000 MWe.^(3.2)

The upper limit on practical size of a single D₂O reactor has not been firmly established but appears from preliminary feasibility studies to be beyond 2000 MWe. The use of pressure tubes in this concept circumvents the problem of fabricating very large pressure vessels and thereby eliminates a major obstacle to scale-up.

3.2 Prototype Reactor

One of the main objectives of a prototype D₂O reactor is to provide a firm basis for determining the construction and operating costs of a reactor of commercial size. To meet this objective, it is essential not only that conditions contemplated for individual fuel positions of a large reactor be demonstrated in the prototype, but also that the prototype contain enough fuel positions for a convincing demonstration of (a) ripple factors caused by fuel assemblies that are operating at different exposure levels, (b) control rod manipulations required for radial flattening of the neutron flux, and (c) fuel management techniques. Of these three items, the control rod manipulation appears to require the largest core size. The minimum size that should be considered is a core with about 160 fuel positions -- half of which would be in a region of radially flattened flux. For a tubular UO₂ assembly of the design contemplated for a full-scale reactor, it is estimated that about 300 MWe can be realized with a 156-position

prototype. Figures 3-2 and 3-3 show the arrangement of the reactor core and an elevation of the reactor building for a 300-MWe prototype reactor plant.

3.3 Engineering Design Studies

The Du Pont Engineering Department is making engineering design studies of D₂O-cooled reactors of three sizes: 300 MWe, 1000 MWe, and 2400 MWe. The features of these reactors are compared in Table 3-I. The 1000-MWe reactor is representative of a full-scale unit that could be placed in operation in the early 1970's after successful operation of the 300-MWe prototype; the 2400-MWe unit is thought to be near the upper limit of economic feasibility for D₂O reactors.

The design of the prototype is being considered in sufficient detail for preparation of a budget estimate of construction cost. We regard a budget estimate as one that is adequate for budgeting of reactor construction funds. This estimate, including turbine plant cost, will be completed by the end of the current fiscal year. The 1000-MWe reactor is to be studied in similar detail; an estimate for this reactor will be completed late in calendar year 1964.

The study of the 2400-MWe reactor is concerned exclusively with the feasibility of scale-up to this size range, and is confined to investigating the major design problems associated with scale-up. These include structural design of the reactor, coolant piping arrangement, provision for on-line refueling, and the number and arrangement of the primary coolant loops. This study is approaching completion; no cost estimates are contemplated. The tentative conclusion is that although such large reactors pose substantial design problems, there is no reason to doubt their engineering feasibility. The major immediate problem appears to be the size limits on individual coolant loops. A 2400-MWe reactor would require ten loops of the largest size that could be available in the near future.

TABLE 3-I

Reactor Design Studies

	<u>Prototype</u>	<u>Reference</u>	<u>Large Scale-up</u>
Electric power, MWe	300	1000	2400
Reactor power, MW	1100	3500	8300
No. of pressure tubes	156	516	1258
Core diameter, ft	16	22	34.5
Core height, ft	15	15	15
No. of control positions	21	37	89
Total D ₂ O flow, 10 ³ gpm	100	350	830
No. of coolant loops	4	6	10
Uranium inventory, tonnes	20	70	150
D ₂ O inventory, tonnes	232	616	1520
Containment sphere diameter, ft	190	250	350
Type of refueling	Shutdown	On-line	On-line
Status	Design & cost study	Design & cost study	Feasibility study

FIG. 3-1 EFFECT OF PLANT SIZE ON ENERGY COST

FIG. 3-2 300 MWe PROTOTYPE REACTOR: REACTOR ARRANGEMENT

FIG. 3-3 300-MWe PROTOTYPE REACTOR: REACTOR BUILDING SECTION

4. USE OF H₂O COOLANT

The use of H₂O rather than D₂O coolant in heavy-water-moderated reactors appears to be technically feasible, but has not been studied in detail.^(4.1) The advantages are a decrease of 40% in the D₂O inventory and a large (perhaps 90%) reduction in D₂O losses because all of the D₂O moderator is confined to low pressure systems.

The principal disadvantage of H₂O cooling is the positive reactivity coefficient associated with an increase in coolant temperature following a power increase. The minimum requirement for stability is that this positive coefficient be counterbalanced by the negative Doppler coefficient associated with the fuel temperature.

Other disadvantages of H₂O coolant include:

- (1) higher fuel enrichment, burnup, and inventory charges;
- (2) a 15% loss in conversion ratio, which makes this concept unsuitable for breeding with thorium fuel; and
- (3) the loss of the potential for use of uranium or thorium metal fuels, since these low-temperature fuels have too small a Doppler coefficient.

A preliminary design of a fuel assembly for use with H₂O coolant is compared in Table 4-I with a D₂O-cooled fuel assembly which can be housed in the same pressure tube. The coolant channel thickness has been reduced by about 35% to reduce the positive component of the reactivity coefficient.

Substitution of H₂O for D₂O and modification of the fuel assembly as indicated would (1) increase the primary coolant pumping power by 50%, (2) decrease the thickness of the coolant annuli, (3) increase the coolant velocity in the fuel from 45 to 60 ft/sec, and (4) increase the fuel enrichment required to give the same fuel exposure from 1.2 to 1.4% U²³⁵. Preliminary estimates indicate that the energy cost from H₂O-cooled and D₂O-cooled reactors are approximately the same if D₂O losses in a D₂O-cooled plant can be held to 2-3% of the inventory. The savings in D₂O cost in a H₂O-cooled reactor are nearly counterbalanced by the higher cost of the uranium and the increased pumping costs.

The minimum technical program to establish definitely the feasibility of this mode of operation would include:

- (1) flow and heat transfer tests at the higher coolant velocity,
- (2) development of suitable spacing devices for the narrower coolant channels,
- (3) physics measurements to establish the values of the coolant and Doppler temperature coefficients of reactivity, and
- (4) an analysis of the safety and control problems associated with operation when the effective reactivity coefficient is near zero.

TABLE 4-I

Comparison of D₂O- and H₂O-Cooled
Fuel Assembly Designs

	<u>D₂O-Cooled Design</u>	<u>H₂O-Cooled Design</u>
Coolant area, in. ²	6.77	5.66
Coolant-to-fuel area	1.12	0.94
Housing tube ID, in.	4.04	4.04
Fuel tubes		
Clad Diameters		
No. 1 OD, in.	3.68	3.80
ID, in.	3.16	3.39
No. 2 OD, in.	2.56	3.01
ID, in.	1.92	2.45
No. 3 OD, in.	1.28	2.05
ID, in.	0.61	1.48
Filler rod OD, in.	-	1.20
Average cladding thickness, in.	0.025	0.025

5. FUEL CYCLE TECHNOLOGY

5.1 Fuel Cycle Characteristics

Table 5-I illustrates the fuel cycle characteristics for a D₂O-cooled reactor operating on slightly enriched uranium. It is seen that the required enrichment is lower than for other converter reactors. This means lower U²³⁵ burnup costs. The combined effect of low enrichment and high specific power leads to low fuel inventory costs. For example, with toll enrichment of U²³⁵ and U₃O₈ at \$5/lb, the value of the U²³⁵ inventory for a 1000-MWe reactor is only about \$3,500,000. At \$40/kg for fabrication, the total value of the fuel assemblies in the reactor is about \$6,200,000, including the uranium cost. This number is several-fold lower than is commonly experienced in power reactors today and offsets the cost of the D₂O inventory.

The U²³⁵ content of spent fuel from D₂O reactors is so low that it is not worth recovering. Therefore, the spent fuel is reprocessed only for recovery of plutonium. The estimated worth of plutonium for power reactor fuel is little more than the cost of recovering it in present-day reprocessing plants. For this reason, a "throwaway" cycle with no spent fuel reprocessing may be attractive for D₂O reactors--particularly if current estimates of fuel reprocessing costs prove to be too optimistic.

Natural uranium cycles are also possible in D₂O reactors, although they appear somewhat more costly at present-day uranium prices. Uranium metal is the best candidate for a natural uranium cycle because its nuclear reactivity is higher than that of oxide. As compared to enriched operation, burnup and fuel inventory charges are lower for the natural uranium cycle, but fabrication expense is higher because fuel exposures are lower.

5.2 Conceptual Design of Fuel Assemblies

The leading candidate fuel concepts for D₂O-cooled reactors are bundles of uranium oxide rods and nested tubes of uranium oxide or uranium metal. The technology of oxide rod bundles is well developed, as attested by their use in the PRTR, CVTR, and NPD, and by their intended use in the CANDU. However, we believe that the nested-tube concept offers better economics

than rod bundles because predicted fabrication costs are lower and power ratings are higher for nested tubes.

Figure 5-1 illustrates the conceptual design of the tubular fuel assemblies Du Pont is developing for D_2O -cooled reactors. Superficially, the design is the same for uranium metal as for uranium oxide. In both instances, the fuel assembly comprises three coaxial tubes spaced from each other and from a Zircaloy housing by spacing ribs. Each tube is cooled on both surfaces. The largest tube is about 3.5 inches in OD, and the smallest is about 1.3 inches in OD; the tubes can be made the same length as the reactor core or can be subdivided into shorter pieces as desired for a particular fuel management scheme. The thickness of the Zircaloy cladding is 0.025 inch.

5.3 Fabrication Processes

Except for spacing devices, complete fabrication processes for uranium oxide tubes and uranium metal tubes have been demonstrated on a pilot scale. More than 100 tubes of each type have been fabricated for irradiation tests by these processes.

Table 5-II summarizes the fabrication steps. Both the oxide and metal processes comprise relatively simple unit operations that are amenable to low-cost production. Oxide tubes are prepared by mechanical compaction. Crushed, fused UO_2 of selected particle sizes is loaded into Zircaloy sheath tubes that are fitted with temporary end plugs, is compacted by vibration to a density of about 85% of theoretical, and then is further densified to at least 90% of theoretical by swaging. The tube is next cut to desired length and counterbored, and permanent end plugs with chambers to hold gases released during irradiation are welded to the sheath tubes.

Fabrication techniques have been developed for UO_2 tubes up to 2.5 inches in diameter; currently, techniques are being developed to fabricate UO_2 tubes corresponding to the outer tube in the three tube designs.^(5.1) These large diameter tubes are scheduled to begin irradiation in May.

The uranium metal tubes are fabricated by a co-extrusion process in which billets of uranium and zirconium are extruded directly to finished cross section dimensions in a single step. End sealing can be accomplished either during the extrusion or by brazing after extrusion.

Electron beam welding appears to be a satisfactory method for attaching spacing ribs to the fuel tube. Although no irradiation tests are yet underway, high quality ribs have been welded to empty sheath tubing and to short coextruded uranium tubes. No difficulty is foreseen in adapting the process to oxide tubes. Alternatively, spacing can be accomplished by welding short spacers at appropriate intervals along the fuel tubes.

5.4 Fabrication Costs

The estimated cost of fabricating UO_2 tubes in the near term is \$40/kg U at throughputs required for a single 1000-MWe reactor. This is exclusive of uranium cost. Roughly 2/3 of this amount represents the cost of preparing crushed, fused UO_2 and the cost of the Zircaloy sheathing. A typical breakdown of the estimated fabrication costs for this size reactor is shown in Table 5-III. A projected cost estimate for a fourfold increase in the production rate results in savings of \$11/kg U with the savings distributed as follows:

- reduction in cost of raw materials, about 60%
- more efficient use of facilities, about 40%

There is insufficient manufacturing experience to predict accurately the fabrication costs for uranium metal tubes of primary interest. However, experience within the AEC suggests that the ultimate fabrication cost for large volume production will be in the neighborhood of \$15/kg. A basic cost advantage of metal tubes is the low cost of Zircaloy components. In contrast to UO_2 tubes and rod bundles, the metal tubes do not require the fabrication of Zircaloy into thin-walled sheath tubing. Instead, the Zircaloy is supplied as relatively inexpensive thick-walled tubing and is formed to size during the coextrusion step. Offsetting this advantage is the lack of any commercial source of low-cost uranium ingot. It is likely that uranium tube costs in the near future will depend strongly on the availability of uranium from AEC production facilities.

5.5 Irradiation Experience

Table 5-IV summarizes the irradiation experience to date with tubular assemblies of uranium oxide and uranium metal. Approximately 150 tubular specimens have been irradiated in either the HWCTR or in the Savannah River production reactors.

Uranium Dioxide

UO₂ tubes have been irradiated at thermal ratings ($\int kd\theta$) of 5 to 73 watts/cm and exposures up to 13,000 MWD/tonne; representative tests performed under power reactor conditions are shown in Table 5-IV. Tubes exposed to 3500 MWD/tonne have been destructively examined, and tubes exposed to 10,000 MWD/tonne are now undergoing examination. The results from these tests indicate that UO₂ tubes should operate satisfactorily in D₂O-cooled reactors at maximum $\int kd\theta$'s of 30-35 watts/cm to exposures of at least 20,000 MWD/tonne.

Problem areas that were revealed during these tests are described below; methods of circumventing these problems have been devised and demonstrated.^(5.2)

Hydride Embrittlement of Zircaloy Cladding. Zircaloy sheath failures due to hydride embrittlement were encountered in early irradiations of UO₂ tubes.^(5.3) Figure 5-2 shows that the sheaths were embrittled from an internal source of hydrogen. A drying step that was incorporated into the fabrication process proved to be effective in eliminating the hydrogen and the resulting hydride embrittlement.^(5.4,5.5) There have been no failures attributed to hydriding of the sheath in 68 oxide tube irradiations since the drying step was adopted; 24 of these tubes were sectioned after irradiation, and in all tubes the Zircaloy sheaths were free of the excessive hydriding that characterized the earlier failures.

Outer Sheath Collapse. During irradiation at high thermal ratings four UO₂ tubes failed by collapse of the outer sheath to form longitudinal wrinkles.^(5.6) In two cases the sheath cracked at the apex of the wrinkle. During fabrication the failed elements were compacted by vibration alone to 82-87% of theoretical oxide density; they were operated at $\int kd\theta$'s of 40-50 watts/cm. Two elements with swage-compacted cores (89-90% of theoretical density) did not fail during irradiation at similar ratings. Cross sections of the two swaged tubes and of two vibrated tubes that failed are shown in Figure 5-3. The failure of the vibratory-compacted tubes was attributed to the large amount of porosity in the core and to the plasticity of the UO₂ at the operating temperature in these elements. At these temperatures the 1200-psi coolant pressure caused appreciable densification of the low density cores and led to collapse and wrinkling of the thin-wall sheathing. The deformation observed in highly rated tubes compacted to different densities is shown in Figure 5-4.

It is concluded that collapse of the outer sheath will not occur in UO_2 tubes swaged to at least 90% of theoretical density and irradiated at thermal ratings of 30-35 watts/cm. This conclusion is based on

- (a) satisfactory performance of swaged tubes at the relatively high thermal ratings of 40-50 watts/cm and exposures to 3500 MWD/tonne, and
- (b) satisfactory performance of both swaged and vibrated tubes (84-90% of theoretical density) at thermal ratings of 25-30 watts/cm.

Inner Sheath Collapse. The only irradiation failure of an oxide tube by collapse of the inner sheath occurred in a Savannah River production reactor at low coolant pressure (Figure 5-5). The failed element contained an unusually large amount of fission and sorbed gas (67 cc/kg UO_2 of fission gas and 118 cc/kg UO_2 of nitrogen), and the columnar grain structure of the core showed that the element had operated at an unusually high central temperature. The reason for the higher-than-expected operating temperature is not known; however, it is believed that the ability of the element to remove heat was impaired by evolved gas, which slightly displaced the sheaths away from the core interface. The resulting increase in central temperature led to the release of additional gas. This process continued until the internal pressure was sufficient to collapse the inner sheath.

It is reasonable to expect that oxide fuel tubes fabricated by present techniques and operated at 30-35 watts/cm can reach burnups of 20,000 MWD/tonne without collapse of the inner sheath. This prediction is supported as follows:

- (a) The current outgassing operation during preparation of the UO_2 effectively removes the sorbed gases. The greatest release of nitrogen from UO_2 tubes irradiated at ratings up to 50 watts/cm was 23 cc/kg UO_2 , which is one-fifth of that observed in the failed element described above (fabricated with UO_2 outgassed under less effective conditions).
- (b) The internal pressure must exceed the reactor operating pressure of 1500-2000 psi before a mechanism that depends on displacement of the inner sheath to impair the heat removing capability of the element becomes operative.

(c) An oxide tube that is compacted to about 90% of theoretical density and that operates at 30 watts/cm will release about 20% of the fission gases formed during irradiation (Figure 5-6). Under these conditions the calculated internal pressure during operation is about 900 psi at a burnup of 20,000 MWD/tonne; this estimate is based on the following assumptions:

- (1) a central temperature of 1600°C and an oxide surface temperature of 400°C throughout the operating life of the element,
- (2) no axial flux gradient,
- (3) no gas retention chambers at the ends of the element, and
- (4) 50 cc/kg UO_2 of gas initially present or desorbed during irradiation.

For a slightly enriched tube operating in a full-scale power reactor, the internal pressure will be even less because the axial flux gradient reduces the average temperature and exposure in the element and because gas retention chambers can be fabricated to provide additional free volume in the element.

Uranium Metal

To achieve attractive fuel cycle costs in D_2O -cooled reactors, uranium metal tubes should be capable of exposure levels of 10,000 MWD/tonne at a metal temperature of about 500°C. Results of initial irradiation tests indicate that unalloyed uranium is probably not sufficiently stable to operate at these conditions, but that uranium alloys under external pressurization have good prospects of attaining them. Two tubes of Zircaloy-clad unalloyed uranium were irradiated at 500°C to 7000 MWD/tonne in the HWCTR, but the test was terminated because of unacceptably high changes in outer diameter in the regions of highest metal temperature.^(5,7) Sticking of the inner housings of the tubes suggested that extensive buckling of the inside diameter of the tubes had occurred. Irradiation of another unalloyed uranium tube with extra thick Zircaloy cladding (0.060 inch) is proceeding in the HWCTR to evaluate the effect of increased external restraints.

Tubes of Zircaloy-clad uranium alloy containing 1.5% molybdenum^(5.8) or 0.1-0.2% of iron and aluminum were irradiated at high metal temperatures in lead-containing capsules and reached high exposures without excessive dimensional changes. The capsules provided external restraints on the specimens to simulate power reactor pressures. The tubes suffered severe core cracking in some instances, and mechanical integrity at high exposures remains to be demonstrated. Tubes of uranium alloy containing iron-aluminum or iron-aluminum-silicon have reached 3400 MWD/tonne at 470°C maximum temperature in the HWCTR with little change in dimension. These tests are continuing, and tests of the uranium-molybdenum alloy have been initiated.

TABLE 5-I

Fuel Cycle for
1000-MWe D₂O-Cooled Reactor

Fuel material and shape	Nested UO ₂ tubes	
Initial enrichment, wt % U ²³⁵	1.2	
Average exposure, MWD/tonne U	15,000	
Fuel inventory, tonnes U	68	
U ²³⁵ inventory, kg	820	
Average specific power, MW/tonne U	54	
Spent fuel assay, g/kg U	<u>Buckled Zone</u>	<u>Flat Zone</u>
U ²³⁵	3.0	1.3
Pu ²³⁹ + Pu ²⁴¹	3.2	3.4
Pu ²⁴⁰	1.3	2.0
Fuel cycle costs, mills/kwh		
U ²³⁵ burnup (\$50/kg)	0.51	
Fabrication (\$40/kg)	0.41	
Reprocessing and shipping	0.29	
Pu credit (\$8.6/g)	(0.29)	
U ²³⁵ inventory at 12.5%	<u>0.11</u>	
Total	1.0	

TABLE 5-II

Fuel Fabrication Processes

Uranium Oxide Tubes

Prepare crushed fused UO_2
Outgas UO_2
Weld spacing ribs on inner sheath tubes^(a)
Load UO_2 into sheath tubes
Vibratory compact to 85% density
Swage to 90-92% density
Weld end plugs with gas chambers

Uranium Metal Tubes

Assemble extrusion billet from machined
uranium and zirconium components
Extrude billet to final cross section
Heat treat
Attach end closures by brazing
Weld spacing ribs

(a) Alternatively, rib can be welded to outer
sheaths after swaging

TABLE 5-III

Estimate of Fabrication Costs

Fuel Description: 3 nested UO_2 tubes, 7-1/2 feet long, 1.3 to 3.7 inches diameter, containing 20 pounds of UO_2 /foot, clad with Zircaloy, spaced with integral ribs.

Throughput: 67 tonnes U per year (1000-MWe reactor).

Plant Operation: One shift operation, 85% acceptance, recycle UO_2 in rejected tubes.

Cost per Assembly
(7-1/2 feet long), \$

Direct Costs

Materials

Zircaloy sheaths (at 23-27 \$/lb)	837
Zircaloy plugs	87
UO_2 (at \$3.4/lb conversion from UF_6)	540
Other	30

Labor and supervision	240
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Other direct charges (maintenance, services, etc)	<u>138</u>
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Total Direct Costs	1872
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Indirect Costs

Indirect plant cost (at 14.46%/yr of plant cost)	270
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Overhead (at 2/3 of direct cost exclusive of materials)	253
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Inventory (at 6%/yr of average value of 2 weeks supply)	<u>3</u>
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Total Indirect Costs	526
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Total Manufacturing Cost per Assembly	2398
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Manufacturing Cost per lb UO_2	16
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Manufacturing Cost per kg U	40
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TABLE 5-IV

Fuel Irradiation Experience

All tests at 1200 psi and cladding surface temperature of 250-300°C

<u>Fuel Type</u>	<u>Max fkdθ, watts/cm</u>	<u>Max Temp, °C</u>	<u>Max Exposure, MWD/tonne U</u>	<u>Status</u>
UO ₂ tubes (2.1-inch OD, 0.030-inch cladding)				
Short tubes	25	-	12,800	Continuing
Short tubes	29	-	10,000	Completed; being inspected
Short tubes	30	-	7,200	Continuing
Long tubes	25	-	10,400	Continuing
Long tubes	30	-	6,300	Continuing
Short tubes	50	-	3,500	Failed; inspected
Short tubes	55	-	1,700	Failed; being inspected
Short tubes	73	-	1,100	Failed; being inspected
Short UO ₂ tubes (2.5-inch OD, 0.020-inch cladding)	23	-	1,000	Continuing
Long unalloyed uranium tubes (2.1-inch OD, 0.025-inch cladding)	-	500	6,800	Completed; being inspected
Short uranium alloy tubes (~2.0-inch OD, 0.025-inch cladding)	-	470	3,400	Continuing

Zr-2.5% Nb Pressure Tube

Zircaloy calandria tube, housing tube,
ribs, and cladding

UO₂ Fuel

FIG. 5-1 FUEL ASSEMBLY FOR D₂O-COOLED-AND-MODERATED REACTOR

FIG. 5-2 CRACKS IN HYDRIDED ZIRCALLOY AFTER IRRADIATION (Mag. 400X)

Neg. 52701

2X

Z-257A, Swaged
 $\int kd\theta \sim 37$ watts/cm
 Large columnar grains.
 Perfect circumferential crack.

Neg. 52358

2X

Z-253C, Vibrated
 $\int kd\theta \sim 41$ watts/cm
 Long radial columnar grains
 very clear because of the
 fractured nature of the cut
 surface.

Neg. 52663

2X

Z-255B, Swaged
 $\int kd\theta \sim 45$ watts/cm
 Long radial columnar
 grains.

Neg. 52662

2X

Z-251C, Vibrated
 $\int kd\theta \sim 49$ watts/cm
 Long radial columnar
 grains.

FIG. 5-3 UO₂ CORE APPEARANCE vs. THERMAL RATING
 AT 37 TO 49 WATTS/CM

FIG. 5-4 EFFECT OF UO_2 DENSITY ON THE COLLAPSE OF HIGHLY RATED OXIDE TUBES

a. Before deformation (nominal)

b. Silhouette of bulges

1X

c. First oblique view

1X

d. Second oblique view

1X

FIG. 5-5 IRRADIATED UO_2 TUBE WITH COLLAPSED INNER SHEATH

FIG. 5-6 FISSION GAS RELEASE IN UO_2 TUBES

6. STATUS OF PHYSICS DEVELOPMENT

6.1 Lattice Physics

The lattice physics of uranium-D₂O systems is on a very firm experimental basis.^(6.1-6.6) Measurements of lattice reactivity have been made on 500 D₂O-moderated lattices in critical and exponential facilities at the Savannah River Laboratory. The ranges of parameters and fuel types included in these studies are shown in Table 6-I. Data are available for rod clusters and tubes of natural uranium oxide, and for rod clusters, tubes, and plate bundles of uranium metal. Typical sets of data are shown in Table 6-II for lattices of UO₂ rod clusters, in Table 6-III for lattices of UO₂ tubes, and in Table 6-IV for lattices of uranium metal tubes. Heavy water temperatures as high as 220°C have been investigated. The Savannah River measurements are supplemented by experimental data obtained elsewhere in the U. S. and in Canada, Sweden, Norway, and France.

A number of calculational codes for uranium-D₂O systems have been normalized to the experimental data and are designed for interpolation between the experimental points.^(6.7-6.9) Codes also have been developed for detailed examination of specific aspects of D₂O reactor systems.^(6.10-6.12) Good results have been obtained in adapting light water calculation methods to the D₂O reactors.

The long-term reactivity predictions for D₂O reactors are supported by measurements of reactivity and isotopic concentrations in slugs of natural uranium metal irradiated to 1000 and 3000 MWD/tonne. The close agreement between observed values and theoretical predictions of both the isotopic concentrations and neutron cross sections promotes confidence in the predictions of burnup effects at even higher exposures.

6.2 Operating Coefficients and Reactor Stability

Zero-power measurements have been made of operating coefficients of reactivity for various uranium-D₂O lattices. Significant data are also being obtained now from operating reactors. The net power coefficient of reactivity for D₂O-cooled reactors is negative; as a result, they are inherently self-stabilizing. After a step insertion of reactivity, a

new steady power level is obtained without need for control rod motion. For UO_2 fuel, power changes are slow and overshoots are small as a consequence of the large time lag in temperature response of the fuel (high heat capacity, low thermal conductivity).

6.3 Fuel Management

The attainment of low fuel cycle costs requires the use of some kind of refueling schedule that will lead to higher average fuel exposures than simple batch discharge of the fuel. Techniques such as countercurrent refueling or radial-shift refueling offer calculated average burnups as high as three times that obtained with simple batch refueling. Calculations have been made of both on-power refueling and shutdown refueling, but no technique other than simple batch refueling has been demonstrated except in the purely mechanical sense. Most of the attention in the U. S. has been devoted to shutdown refueling. One technique that appears acceptable is a two-zone refueling plan in which fresh fuel assemblies are charged at different frequencies to a central flat zone and to a peripheral buckled zone. Each fuel assembly remains in one position during its entire stay in the reactor. Preliminary calculations indicate that this procedure affords a satisfactory compromise between radial power distribution and core life.

TABLE 6-I

Surveys of D₂O-Uranium Lattices

Rod clusters of natural uranium oxide

Lattice pitch

Triangular	7.0 to 16.2 in.
Square	8.0 in.

Moderator-to-fuel (UO ₂) volume ratio	8.8 to 44.4
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Tubes of natural uranium oxide

No. of tubes per assembly	2, 3, and 4
Lattice pitch - triangular	9.3 and 11.1 in.
Moderator-to-fuel (UO ₂) volume ratio	9.2 to 17.3

Rod clusters of natural uranium metal

Lattice pitch

Triangular	7.0 to 21.0 in.
Square	1.5 to 4.0 in.

Moderator-to-fuel volume ratio	7.7 to 161.5
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Tubes of natural uranium metal

No. of tubes per assembly	1, 2, and 4
Lattice pitch - triangular	7.0 to 14.0 in.
Moderator-to-fuel volume ratio	12.8 to 59.8

Clusters of natural uranium metal plates

Lattice pitch - triangular	7.0 to 10.7 in.
Moderator-to-fuel volume ratio	8.9 to 61.2

TABLE 6-II

Buckling Measurements for Uranium Oxide Rod Clusters:
Single-Region Experiments

D_2O Temp = $20^\circ C$

Lattice Number	No. of Rods per Assembly	Lattice Pitch, inches	OD of Housing Tube, inches	D_2O or Air in Housing Tube	Radial Buckling (B_r^2), μB	Vertical Buckling (B_z^2), μB
1	19	8.08	None	-	337.9 $\pm$ 5.2	241.5 $\pm$ 2.5
2		9.33	None	-	351.0 $\pm$ 4.6	217.3 $\pm$ 3.7
3		9.33	4	D_2O	212.1 $\pm$ 3.0	271.3 $\pm$ 3.0
4		9.33	4	Air	212.6 $\pm$ 1.4	221.2 $\pm$ 2.0
5		9.33	5	D_2O	213.0 $\pm$ 1.5	237.0 $\pm$ 3.4
6		9.33	5	Air	176.1 $\pm$ 0.6	138.0 $\pm$ 1.4
7		12.12	None	-	217.8 $\pm$ 1.4	171.7 $\pm$ 0.9
8		12.12	4	D_2O	217.1 $\pm$ 2.6	127.4 $\pm$ 0.4
9		12.12	4	Air	213.8 $\pm$ 1.8	143.1 $\pm$ 1.2
10		14.00	None	-	153.4 $\pm$ 1.3	127.0 $\pm$ 0.6
11	31	9.33	None	-	207.6 $\pm$ 3.6	304.5 $\pm$ 2.7(a)
12		9.33	5	D_2O	210.3 $\pm$ 2.6	212.6 $\pm$ 1.7
13		9.33	5	Air	201.5 $\pm$ 4.0	120.8 $\pm$ 1.2
14		11.10	None	-	243.5 $\pm$ 1.1	252.5 $\pm$ 2.8
15		12.12	None	-	216.9 $\pm$ 3.4	252.1 $\pm$ 0.8(a)
16		12.12	5	D_2O	216.0 $\pm$ 2.1	182.9 $\pm$ 2.4
17		12.12	5	Air	212.1 $\pm$ 2.8	193.8 $\pm$ 1.1
18		14.00	None	-	221.9 $\pm$ 1.7	131.7 $\pm$ 0.8
19		16.17	None	-	87.6 $\pm$ 1.0	175.7 $\pm$ 1.6
20		12.12	None	-	214.4 $\pm$ 2.0	247.5 $\pm$ 2.4
21	48	12.12	6	D_2O	215.9 $\pm$ 0.6	180.7 $\pm$ 2.0
22		12.12	6	Air	206.8 $\pm$ 1.3	171.9 $\pm$ 0.9
23		14.00	None	-	152.2 $\pm$ 0.8	256.6 $\pm$ 3.8
24		14.00	6	D_2O	151.5 $\pm$ 3.2	201.8 $\pm$ 1.3
25		14.00	6	Air	147.9 $\pm$ 1.0	240.4 $\pm$ 2.6

(a) Moderator purity = 99.68 mol % in these lattices. In all other lattices the moderator purity was 99.59 mol % D_2O .

TABLE 6-III

Physics Parameters of Tubular UO₂ Fuel Elements

(From Reference 6.7)

A. Description of Fuel and Shroud Tubes

Tube Designation	Clad* ID, in.	UO ₂ ID, in.	UO ₂ OD, in.	Clad* OD, in.	UO ₂ Density, g/cm ³
A	0.50	0.56	1.06	1.12	8.62
B'	1.18	1.24	2.04	2.10	8.70
B	1.48	1.54	2.04	2.10	8.72
C	2.46	2.52	3.02	3.08	8.72
C'	2.46	2.52	3.32	3.38	8.59
D	3.44	3.50	4.00	4.06	8.72
E'†	3.57			3.75	
E†	4.25			4.50	
F†‡	0.81			1.00	

*The clad is type 1100 aluminum alloy.

†The E', E, and F tubes are made from 6061 aluminum alloy.

‡The F tube is sealed so that it always remains voided.

Table 6-III B. PLATR Measurements and Calculated Values of k_{∞} and f for Tubular UO_2 Fuel Elements

Exp. No.	Fuel* Element	Coolant Material	Density g/cm ³	Coolant Channels Filled	Lattice Pitch, in.	Moderator Purity, % D ₂ O	Measured			Calculated	
							f	k_{∞}	k_{∞} (99.75% D ₂ O)	f	k_{∞}
1	C	D ₂ O	1.105	Central	7.50	99.35	0.945	1.180	1.196	0.946	1.162
2	C	D ₂ O	1.105	None	7.50	99.35		1.174†			
3	C	D ₂ O	1.105	Central	10.46	99.60	0.930	1.178	1.192	0.932	1.181
4	C	D ₂ O	1.105	None	10.46	99.59	0.931	1.184	1.198	0.933	1.194
5	C'	D ₂ O	1.105	Central	7.50	99.32	0.964	1.132	1.141	0.964	1.128
6	C'	D ₂ O	1.105	None	7.50	99.32		1.128†			
7	FB'C'E'	D ₂ O	1.105	All	7.50	99.38	0.946	1.058	1.060	0.944	1.053
8	FB'C'E'	D ₂ O	1.105	FB'+B'C'	7.50	99.38		1.057†			
9	FB'C'E'	D ₂ O	1.105	FB'+C'E'	7.50	99.38		1.065†			
10	FB'C'E'	D ₂ O	1.105	B'C'+C'E'	7.50	99.38		1.060†			
11	FB'C'E'	D ₂ O	1.105	None	7.50	99.38	0.946	1.062	1.063	0.946	1.066
12	FB'C'E'	D ₂ O	1.105	All	8.46	99.53	0.943	1.097	1.100	0.943	1.100
13	FB'C'E'	D ₂ O	1.105	FB'+B'C'	8.46	99.53		1.099†			1.105
14	FB'C'E'	D ₂ O	1.105	FB'+C'E'	8.46	99.53		1.104†			
15	FB'C'E'	D ₂ O	1.105	B'C'+C'E'	8.46	99.53		1.099†			
16	FB'C'E'	D ₂ O	1.105	None	8.46	99.53	0.944	1.106	1.109	0.944	1.114
17	FB'C'E'	H ₂ O	0.08	All	8.46	99.53		1.100†			1.104
18	FB'C'E'	H ₂ O	0.08	FB'+B'C'	8.46	99.53		1.103†			
19	FB'C'E'	H ₂ O	0.08	C'E'	8.46	99.53		1.104†			
20	FB'C'E'	H ₂ O	0.16	All	8.46	99.53		1.095†			
21	FB'C'E'	H ₂ O	0.16	FB'+B'C'	8.46	99.53		1.100†			
22	FB'C'E'	H ₂ O	0.16	C'E'	8.46	99.53		1.101†			
23	FB'C'E'	H ₂ O	0.28	All	8.46	99.53		1.087†			1.086
24	FB'C'E'	H ₂ O	0.28	FB'+B'C'	8.46	99.53		1.095†			
25	FB'C'E'	H ₂ O	0.28	C'E'	8.46	99.53		1.098†			
26	FB'C'E'	H ₂ O	0.4	All	8.46	99.53		1.079†			
27	FB'C'E'	H ₂ O	0.4	FB'+B'C'	8.46	99.53		1.091†			
28	FB'C'E'	H ₂ O	0.4	C'E'	8.46	99.53		1.094†			
29	FB'C'E'	H ₂ O	0.5	All	8.46	99.53		1.073†			1.070
30	FB'C'E'	H ₂ O	0.5	FB'+B'C'	8.46	99.53		1.087†			
31	FB'C'E'	H ₂ O	0.5	C'E'	8.46	99.53		1.091†			
32	FB'C'E'	Dowtherm-A	1.063	All	8.46	99.53		1.061†			
33	FB'C'E'	Dowtherm-A	1.063	FB'+B'C'	8.46	99.53		1.079†			
34	FB'C'E'	Dowtherm-A	1.063	C'E'	8.46	99.53		1.086†			
35	FB'C'E'	D ₂ O	1.105	All	10.46	99.70	0.939	1.150	1.152	0.940	1.152
36	FB'C'E'	D ₂ O	1.105	FB'+B'C'	10.46	99.70		1.155†			
37	FB'C'E'	D ₂ O	1.105	FB'+C'E'	10.46	99.70		1.158†			
38	FB'C'E'	D ₂ O	1.105	B'C'+C'E'	10.46	99.70		1.152†			
39	FB'C'E'	D ₂ O	1.105	None	10.46	98.74	0.907	1.122	1.159	0.906	1.130
40	FB'C'E'	D ₂ O	1.105	None	10.46	99.71	0.940	1.158	1.159	0.941	1.165
41	FB'C'E'	H ₂ O	0.08	All	10.46	99.71		1.149†			1.156
42	FB'C'E'	H ₂ O	0.08	FB'+B'C'	10.46	99.71	0.940	1.153†			
43	FB'C'E'	H ₂ O	0.08	C'E'	10.46	99.71		1.153†			
44	FB'C'E'	H ₂ O	0.16	All	10.46	99.71		1.140†			
45	FB'C'E'	H ₂ O	0.16	FB'+B'C'	10.46	99.71		1.149†			
46	FB'C'E'	H ₂ O	0.16	C'E'	10.46	99.71		1.148†			
47	FB'C'E'	H ₂ O	0.28	All	10.46	99.71		1.126†			1.134
48	FB'C'E'	H ₂ O	0.28	FB'+B'C'	10.46	99.71		1.140†			
49	FB'C'E'	H ₂ O	0.28	C'E'	10.46	99.71		1.140†			
50	FB'C'E'	H ₂ O	0.4	All	10.46	99.71		1.111†			

†These values of k_{∞} obtained from measurements of change in k_{∞} from dry cluster produced by filling indicated channels.

Table 6-III B (Continued)

Exp. No.	Fuel* Element	Coolant Material	Density, g/cm ³	Coolant Channels Filled	Lattice Pitch, in.	Moderator Purity, % D ₂ O	Measured			Calculated	
							f	k _∞	k _∞ (99.75% D ₂ O)	f	k _∞
51	FB'C'E'	H ₂ O	0.5	All	10.46	99.71		1.100†			1.113
52	FB'C'E'	Dowtherm-A	1.063	All	10.46	99.71		1.083†			1.097
53	FB'C'E'	Dowtherm-A	1.063	FB'+B'C'	10.46	99.71		1.119†			
54	FB'C'E'	Dowtherm-A	1.063	C'E'	10.46	99.71		1.118†			
55	FB'C'E'	D ₂ O	1.105	All	13.22	99.45	0.908	1.130	1.153	0.908	1.144
56	FB'C'E'	D ₂ O	1.105	FB'+B'C'	13.22	99.45		1.136†			
57	FB'C'E'	D ₂ O	1.105	FB'+C'E'	13.22	99.45		1.137†			
58	FB'C'E'	D ₂ O	1.105	B'C'+C'E'	13.22	99.45		1.131†			
59	FB'C'E'	D ₂ O	1.105	None	13.22	99.43	0.908	1.139	1.163	0.909	1.154
60	FB'C'E'	H ₂ O	0.08	All	13.22	99.43		1.130†			
61	FB'C'E'	H ₂ O	0.08	FB'+B'C'	13.22	99.43		1.135†			
62	FB'C'E'	H ₂ O	0.08	C'E'	13.22	99.43		1.134†			1.146
63	FB'C'E'	H ₂ O	0.16	All	13.22	99.43		1.121†			
64	FB'C'E'	H ₂ O	0.16	FB'+B'C'	13.22	99.43		1.130†			
65	FB'C'E'	H ₂ O	0.16	C'E'	13.22	99.43		1.129†			
66	FB'C'E'	H ₂ O	0.28	All	13.22	99.43		1.109†			1.123
67	FB'C'E'	H ₂ O	0.28	FB'+B'C'	13.22	99.43		1.124†			
68	FB'C'E'	H ₂ O	0.28	C'E'	13.22	99.43		1.123†			
69	FB'C'E'	H ₂ O	0.40	All	13.22	99.43		1.096†			
70	FB'C'E'	H ₂ O	0.40	FB'+B'C'	13.22	99.43		1.117†			
71	FB'C'E'	H ₂ O	0.40	C'E'	13.22	99.43		1.113†			
72	FB'C'E'	H ₂ O	0.50	All	13.22	99.43		1.086†			1.102
73	FB'C'E'	H ₂ O	0.50	FB'+B'C'	13.22	99.43		1.112†			
74	FB'C'E'	H ₂ O	0.50	C'E'	13.22	99.43		1.109†			
75	FB'CDE	D ₂ O	1.105	All	10.46	99.37	0.920	1.103	1.113		1.097
76	FB'CDE	D ₂ O	1.105	B'C+CD+DE	10.46	99.37		1.104†			
77	FB'CDE	D ₂ O	1.105	FB'+CD+DE	10.46	99.37		1.108†			
78	FB'CDE	D ₂ O	1.105	FB'+B'C+DE	10.46	99.37		1.111†			
79	FB'CDE	D ₂ O	1.105	FB'+B'C+CD	10.46	99.37		1.106†			
80	FB'CDE	D ₂ O	1.105	None	10.46	99.37		1.120†			
81	ABCDE	D ₂ O	1.105	All	10.46	99.56	0.928	1.101	1.106		1.098
82	ABCDE	D ₂ O	1.105	BC+CD+DE	10.46	99.37		1.105†			
83	ABCDE	D ₂ O	1.105	AB+CD+DE	10.46	99.37		1.108†			
84	ABCDE	D ₂ O	1.105	AB+BC+DE	10.46	99.37		1.111†			
85	ABCDE	D ₂ O	1.105	AB+BC+CD	10.46	99.37		1.105†			
86	ABCDE	D ₂ O	1.105	None	10.46	99.58	0.931	1.124	1.128		1.123
87	ABCDE	H ₂ O	0.08	CD	10.46	99.70		1.118†			
88	ABCDE	H ₂ O	0.08	DE	10.46	99.70		1.120†			
89	ABCDE	H ₂ O	0.08	All	10.46	99.70		1.112†			
90	ABCDE	H ₂ O	0.16	DE	10.46	99.70		1.116†			
91	ABCDE	H ₂ O	0.16	All	10.46	99.70		1.100†			
92	ABCDE	H ₂ O	0.28	CD	10.46	99.70		1.107†			
93	ABCDE	H ₂ O	0.28	DE	10.46	99.70		1.112†			
94	ABCDE	H ₂ O	0.28	All	10.46	99.70		1.085†			
95	ABCDE	H ₂ O	0.40	CD+DE	10.46	99.70		1.083†			
96	ABCDE	H ₂ O	0.40	All	10.46	99.70		1.068†			
97	ABCDE	H ₂ O	0.5	CD	10.46	99.70		1.097†			
98	ABCDE	H ₂ O	0.5	DE	10.46	99.70		1.099†			
99	ABCDE	H ₂ O	0.5	All	10.46	99.70	0.894	1.060	1.063		

†These values of k_∞ obtained from measurements of change in k_∞ from dry cluster produced by filling indicated channels.

TABLE 6-IV

Bucklings of Tubular Assemblies of Uranium Metal in D₂O

Fuel Assembly Designation ^(a)	Lattice Pitch, inches	Fuel Assembly Contains	Measured Bucklings, μB ^(b)			Calculated Bucklings, μB
			B_z^2	B_r^2	B_m^2	B_m^2
"ABC"	9.33	Air	212 $\pm$ 6	378 $\pm$ 5	590 $\pm$ 8	543
		D ₂ O	178 $\pm$ 8	375 $\pm$ 3	553 $\pm$ 9	531
"ABC"	12.12	Air	386 $\pm$ 6	213 $\pm$ 3	599 $\pm$ 7	593
		D ₂ O	330 $\pm$ 7	214 $\pm$ 2	544 $\pm$ 7	539
"ABC"	14.00	Air	341 $\pm$ 6	164 $\pm$ 1	505 $\pm$ 6	500
		D ₂ O	282 $\pm$ 6	166 $\pm$ 2	448 $\pm$ 6	437
"C"	9.33	Air	195 $\pm$ 5	375 $\pm$ 1	570 $\pm$ 5	529
		D ₂ O	286 $\pm$ 8	385 $\pm$ 3	671 $\pm$ 8	655
"C"	12.12	Air	312 $\pm$ 5	212 $\pm$ 3	524 $\pm$ 6	515
		D ₂ O	341 $\pm$ 6	217 $\pm$ 4	558 $\pm$ 7	542
"C"	14.00	Air	270 $\pm$ 4	168 $\pm$ 2	438 $\pm$ 5	426
		D ₂ O	269 $\pm$ 5	171 $\pm$ 2	440 $\pm$ 5	420

(a)	Component	Material	OD, inches	Wall Thickness, inch
"ABC"	Fuel rod	Uranium	1.000	-
	Cladding tube	1100 Aluminum	1.090	0.032
	Inner fuel tube	Uranium	2.120	0.180
"C"	Outer fuel tube	Uranium	3.500	0.320
	Housing tube	6063 Aluminum	4.000	0.050

(b) $1 \mu B = 10^{-6} \text{ cm}^{-2}$

7. MECHANICAL DESIGN

As mentioned previously, the Du Pont Engineering Department is currently engaged in three studies related to mechanical design and cost of power reactors moderated and cooled by liquid D_2O . Sargent & Lundy, Engineers, are assisting in these studies.

7.1 Reactor Description

Some of the more important design features of these reactors are described briefly. The discussion for the most part is restricted to features which differ markedly from pressurized H_2O reactors. Illustrations have been taken from drawings prepared in current studies of a 1000-MWe reactor. Some of these drawings are not yet complete. Figure 7-1 shows a partial process flowsheet for the 1000-MWe plant. A single-pressure-tube reactor is served by six primary coolant loops circulating a total of 350,000 gpm of D_2O through the pressure tubes and fuel at a pressure of 1700 psi and at an average outlet temperature of about $300^{\circ}C$. Each loop contains a single pump and a single vertical steam generator. Saturated steam at 680 psia then flows to a single turbine-generator unit of the largest size currently offered. The steam conditions may not be optimum.

A vertical section of the reactor itself is shown in Figure 7-2. The pressure-tube reactor is particularly suited to heavy water reactors because of the clumping of fuel into widely separated massive fuel elements, which is permissible and desirable with heavy water moderation. This permits reactor scale-up to very large sizes since the reactor vessel itself operates at substantially atmospheric pressure.

The moderator tank is about 25 ft in diameter by 20 ft high and is penetrated by 553 thin-walled calandria tubes on a 10-inch-square pitch. Figure 7-3 shows the reactor lattice arrangement in which 516 tubes serve as pressure tubes for fuel positions and 37 contain clusters of control rods for reactor control and flux shaping. About 55 smaller calandria tubes provide interstitial positions for safety rods and in-core instrumentation.

Pressure tubes which contain fuel with an active fuel length of 15 ft pass through the calandria tubes and are insulated from them by a narrow gas annulus. Coolant is supplied to the pressure tubes and removed from them by individual coolant pipes leading from circular headers above and below the reactor. The arrangement of the primary coolant loops is shown in Figure 7-4.

Moderator is circulated through the reactor tank and control and safety rod positions by auxiliary cooling circuits operating at low temperature and pressure. A simplified flowsheet of the auxiliary cooling system is shown in Figure 7-5.

Substantial axial shields containing H_2O and steel are located above and below the reactor; H_2O is circulated through the shields for cooling. Each shield is independently supported from the building structure, with the bottom shield supporting the moderator tank and the top shield supporting the pressure tubes and fuel charge. A radial thermal shield of steel and H_2O is also provided.

The major design problems for the reactor include the pressure tubes, the large axial shield structures (Du Pont has considerable design experience with such structures), the complex piping at the top and bottom of the reactor to provide for access to the reactor and for thermal expansion of piping, and the support of the heavy reactor structure that cantilevers over the lower coolant header vault.

Vertical and plan sections of the proposed reactor containment building are shown in Figures 7-6 and 7-7. A 250-ft-diameter spherical building appears to be the optimum size both for housing reactor building equipment and for containing the energy and fission products that would be released if fuel meltdown were to follow a major rupture in the primary cooling system. The building would be fabricated from 1-1/4-in.-thick steel plate for an internal design pressure of 25 psig and for conservative external loadings from wind and snow. A deluge system with recirculating cooled water and absorbers for iodine in the air would reduce the temperature, pressure, and fission product content of the building atmosphere after a major accident and thus minimize the effect of any building leakage.

The reactor building is arranged for on-power refueling from beneath the reactor with control rod drives located above the reactor. We do not necessarily advocate on-power refueling but have required that our designs for large reactors be compatible with it.

7.2 Component Development

The problems and status of the required development effort on several components of reactor design merit some discussion. The pressure tube assembly is the basic building block in design of a pressure-tube reactor and thus deserves high priority. Figure 7-8 illustrates a preliminary pressure tube design for the 1000-MWe reactor. Critical features in the design include

the development of reliable joints between the Zircaloy in-core portion of the tube and the stainless or "Inconel" end assemblies, the development of high-strength zirconium alloys to minimize wall thickness, and the development of reliable closures at the ends of the assembly to permit refueling.

Preliminary development of a tandem-extruded Zircaloy - stainless steel joint of the type illustrated has been performed for the AEC by Du Pont and Nuclear Metals, Inc. Sample joints have been subjected to severe thermal cycling tests and to reactor irradiation equivalent to long service life, and have not shown evidence of deterioration.

Pressure tubes at least roughly similar to those required have been developed for D₂O service in the Canadian NPD and CANDU reactors, in the PRTR at Hanford, and in the CVTR at Parr, S. C. Du Pont proposes to proceed with development and thorough testing of prototype assemblies, at least until such work is assumed by others as part of a reactor construction program. Both the internally insulated pressure tube (thin annuli filled with D₂O) and the calandria-type design (with hot pressure tube insulated from the calandria tube by a gas annulus, as already described) will remain under consideration until their relative merits are thoroughly understood.

The reactor fueling machine will require considerable design and development effort. It is clear that on-line refueling offers potential economies both through elimination of reactor downtime for refueling and through maximum average fuel life at minimum enrichment. Because the on-line fueling machine is costly, and because it necessarily adds some potential hazard to reactor operation, it is our present belief that the desirability of on-line refueling will depend on the importance of continuity of reactor operation to the particular utility operating the reactor.

In development of a preliminary design for the prototype reactor, it is our objective to provide an off-line fueling machine which could refuel one-third of the reactor during a weekend shutdown perhaps three times a year. At modest cost in fuel economy it should then be possible to refuel the entire reactor annually with a total downtime of about one week. The reliable high-speed machine that would be required for this service has not been used in the power reactor field. Du Pont has some ten years of successful experience in the use of similar machines, however.

The Canadians have devoted considerable effort to the development of on-line refueling machines for the NPD and CANDU reactors. Within the last few months the NPD machines have been placed in service at full reactor pressure and temperature with apparent success. The American Machine and Foundry Company has been carrying out the development of a machine of alternative design to provide a backup to the Canadian effort.

Figures 7-9 and 7-10 illustrate the application of on-power refueling to the 1000-MWe reactor. Figure 7-9 shows the fueling machine beneath the reactor, with indication of how the fueling snout is extended to engage the end of a pressure tube. New and spent fuel (1/2-length tubes) is stored in a rotating compartmental chamber within the machine. The machine is maintained hot and pressurized at all times. Figure 7-10 illustrates the method of charging new fuel to the machine and of discharging spent fuel to the storage basin. Fuel transfer chambers are pressurized before they are engaged by the fueling machine.

7.3 Factors Which May Limit Reactor Size

Du Pont's feasibility study of the 1000- and 2400-MWe reactors has not revealed any design problems which appear beyond the reach of normal design and development effort, and it has not indicated that construction costs for the larger plant should be expected to be disproportionately high. There are several considerations, however, that indicate that the 2400-MWe plant may possibly be approaching the maximum economic reactor size. The following appear to be critical factors which may hinder achievement of decreased power cost from very large reactors:

1. The increasing complexity of coolant piping above and below the reactor.
2. The increasing number of primary coolant loop pumps and steam generators required with present technology (ten of each in the 2400-MWe design). Economical equipment arrangement within a containment building becomes difficult.
3. Support of the tremendously heavy reactor structure above a coolant header vault.
4. Structural design of the building. Spherical containment buildings much above 350 ft in diameter (2400-MWe case) require stiffening or excessive plate thickness for support against external loads.

7.4 D₂O Losses

The economic penalty of possible excessive rates of heavy water loss during reactor operation or shutdown has been of great concern to those considering construction of heavy water reactors. A D₂O loss rate of 100 lb/day (2%/yr) in a 1000-MWe reactor plant would, for example, impose a penalty of about 0.1 mill/kwh on the cost of electric power generated.

Unfortunately, those D₂O-cooled pressure tube reactors which have accumulated substantial operating time (NPD and PRTR) have been built and operated for test and development purposes and thus provide relatively little assistance in predicting losses from large plants with infrequent shutdown. The best performance we are aware of is a loss rate equivalent to 7%/yr (14 lb/day) of the inventory in a one-month period of HWCTR operation. While it is our belief that the cost of D₂O losses can be held to the level of 0.1 mill/kwh by careful design and thorough development work on seals and closures, this remains to be proven in a prototype plant.

FIG. 7-1 1000-MWe REACTOR: FLOW DIAGRAM OF PRIMARY SYSTEM

FIG. 7-2 1000-MW_e REACTOR: REACTOR ARRANGEMENT

FIG. 7-3 1000-MWe REACTOR: COOLANT HEADER ARRANGEMENT

FIG. 7-4 1000-MWe REACTOR: PRIMARY COOLING LOOP

FIG. 7-5 1000-MWe REACTOR: FLOW DIAGRAM OF AUXILIARY SYSTEMS

FIG. 7-6 1000 MWe REACTOR: REACTOR BUILDING SECTION

FIG. 7-7 1000 MWe REACTOR: REACTOR BUILDING PLAN

FIG. 7-8 1000 MW_e REACTOR: PRESSURE TUBE ASSEMBLY

FIG. 7-9 1000 MWe REACTOR: ON-POWER REFUELING MACHINE

FIG. 7-10 1000 MWe REACTOR: ON-POWER FUEL HANDLING AREA

8. OPERATING EXPERIENCE AND OTHER PROGRAMS

The best indicator of the depth of available experience with D_2O -cooled reactors is the fact that 50 reactor-years of operation has been achieved with such units. This experience includes a total of 5 reactor-years in three different reactors at operating pressures in the range required for electric power generation. A list of some characteristics of the principal D_2O -cooled reactors is presented in Table 8-I.

The broadest experience in D_2O cooling is at Savannah River, where five D_2O -cooled production reactors have been in operation for nearly a decade. Although these reactors operate at low pressure, much of the experience gained in such areas as uranium- D_2O physics, neutron flux management in large, highly rated cores, refueling techniques, D_2O chemistry and handling, and over-all reactor design is applicable to power reactors.

Two pressure-tube, D_2O -cooled-and-moderated reactors (the PRTR at Hanford and the NPD at Rolphton, Ontario) have been in operation at elevated pressure and temperature for more than one year. A third reactor of this type, the CVTR at Parr, S. C., has just been started up. All three of these reactors are contributing direct experience on the operation of pressure-tube, D_2O -cooled power reactors. Similar experience has been gained by almost two years of operation of the HWCTR, a fuel test reactor at Savannah River. This reactor is not of the pressure tube type, but otherwise operates at conditions of interest in power reactor design. The boiling water reactor at Halden, Norway, is in the same category. The NPR at Hanford provides direct experience in the important areas of pressure tube and fuel technology, although it is not a D_2O reactor.

TABLE 8-I

Operating D₂O-Cooled Reactors

	<u>No. of Reactors</u>	<u>Date Critical</u>	<u>Design Thermal Power, MW</u>	<u>Pressure, psi</u>	<u>Fuel</u>	<u>Remarks</u>
Savannah River production reactors	5	1953-5	-	-	-	Low pressure
HBWR (Norway)	1	1959	20	1500	UO ₂ rods	Pressure vessel; boiling D ₂ O coolant
PRTR	1	1961	70	1000	UO ₂ rods and tubes	Pressure tube
CVTR	1	1963	65	1500	UO ₂ rods	Pressure tube
NPD	1	1962	81	1000	UO ₂ rods	Pressure tube
HWCTR	1	1962	70	1200	Various	Pressure vessel
CANDU	1	1965	698	1400	UO ₂ rods	Pressure tube; under con- struction

9. SAVANNAH RIVER FACILITIES FOR REACTOR DEVELOPMENT

Extensive facilities and an experienced technical staff are available at the Savannah River Laboratory for further development of D₂O reactors. In addition to the HWCTR, the principal facilities are the Process Development Pile, a full-scale D₂O-moderated zero-power reactor; the Pressurized Subcritical Experiment, a D₂O-moderated exponential reactor for physics measurements at a temperature of about 225°C; complete experimental fabrication facilities for tubular UO₂ assemblies; heat transfer and flow test loops; and various supporting facilities.

9.1 Heavy Water Components Test Reactor (HWCTR)

The HWCTR^(9.1) (shown in Figure 9-1) is a heavy-water-moderated pressure-vessel reactor designed for irradiating power reactor fuel at coolant temperatures up to 250°C and at pressures up to 1500 psig. One isolated liquid-cooled loop can be operated at 500 psi above the reactor pressure. The reactor vessel (shown in Figure 9-2) has 36 fuel positions; twelve of these are normally used as test positions and the other 24 positions are occupied by drivers. For proof tests of fuel elements in stages of advanced development, all 36 positions can be used as fuel test positions. Neutron flux in the HWCTR is high (10^{14} n/cm²-sec).

The capabilities of the HWCTR are effectively combined with other SRL facilities (fuel fabrication and postirradiation inspection in caves and underwater) to form a complex for power reactor fuel and component development and testing.

9.2 Experimental Physics Facilities

Facilities for studying heavy water lattices include the Process Development Pile (PDP)^(9.2) shown in Figure 9-3, a large critical assembly moderated with 110 tons of D₂O. This reactor is a flexible tool for lattice studies and reactor mockups. A second reactor, the Standard Pile (SP), acts as a neutron source for irradiations and as a sensitive test reactor for reactor safety studies. Its principal use, however, is to feed neutrons into a five-foot-diameter tank of the Subcritical Exponential (SE)^(9.3); this facility is shown in Figure 9-4. The SE is used to study both light and heavy water lattices. The second

exponential, the Pressurized Subcritical Experiment (PSE)^(9.4) is shown in Figure 9-5. This facility is for studying high-temperature D₂O lattices. The PSE source of 10¹⁰ neutrons/second is supplied by antimony-beryllium sources.

9.3 Receiving Basin for Off-Site Fuels (RBOF)

The RBOF^(9.5) provides facilities for storing, disassembling, inspecting, and cutting irradiated fuel and components. This facility is used for interim nondestructive examinations of power reactor fuel being irradiated in the HWCTR. Full-length irradiated fuel tubes can be cut to short lengths in RBOF for detailed metallurgical examination in the High Level Caves. Figure 9-6 is a photograph of the main storage basin in the RBOF.

9.4 High Level Caves

A large High Level Cave facility (shown in Figure 9-7) is available for performing remotely a variety of postirradiation inspections and metallographic examinations of power reactor fuel. The facility includes ten cells with three-foot-thick concrete walls and lead-glass windows. There are four general-purpose manipulators and 26 master-slave manipulators capable of performing numerous operations within the cells.

9.5 Fuel Fabrication Laboratory

The Fuel Fabrication Laboratory covers a floor area of 26,300 square feet. This facility (shown in Figure 9-8) has the capability of pilot-scale production of various shapes of power reactor fuel and control elements. Extensive equipment is available for research and development in the fabrication and evaluation of intricate shapes of almost any metal, powder metal, or ceramic.

9.6 Heat Transfer Laboratory

A Heat Transfer Laboratory has a loop rated for pressures up to 1200 psi. The loop is equipped for operation to 60% exit steam quality and can handle test sections up to 18 feet long. Mobile test sections are available for testing heat transfer limits over a wide range of geometries and operating conditions. The facility may be used to study heat transfer phenomena with liquid, two-phase boiling, and fog cooling.

9.7 Hydraulics Test Facilities

The Hydraulics Laboratory is available for studying the flow characteristics of all test assemblies irradiated in the HWCTR. Facilities in this laboratory include:

- A mockup of one-sixth of a large reactor for studying moderator circulation patterns and for visual observation of component movement in the moderator section of a reactor under various flow conditions.
- Hydraulic loops for long-term flow tests to determine vibration and erosion damage to fuel and components.
- Test stands (shown in Figure 9-9) used to determine flow distribution and to measure total flow through test assemblies before irradiation in the HWCTR.

In addition, flow and temperature monitoring efficiency may be determined for various power reactor fuel designs. Facilities are also available for long-term corrosion studies.

9.8 Computer Program for Evaluation Studies

A computer program is available from the Savannah River Laboratory for performing the lengthy calculations required to make preliminary design and cost estimates for D₂O-moderated reactors. The code is written in FORTRAN language for an IBM-704. The code is applicable to reactor systems with liquid D₂O, boiling D₂O, and organic coolants at power levels of 300, 500, and 1000 MWe. A flowsheet for the computer program is shown in Figure 9-10.

FIG. 9-1 VIEW OF HEAVY WATER COMPONENTS TEST REACTOR

FIG. 9-3 PROCESS DEVELOPMENT PILE

Type:	D ₂ O moderated, tank. No cooling.
Purpose:	Lattice physics
Size:	Tank 16' dia. 15 1/2' high
Power Level:	1 Kw.
Fuel:	Any
Startup:	1953
Features:	Continuously variable lattice arrangement. Flexible operation. Moderator temperature range 10 to 90°C.
Owner/Operator:	USAEC/E. I. du Pont de Nemours and Co.

FIG. 9-4 SUBCRITICAL EXPERIMENT

Type:	Exponential Tank. H ₂ O or D ₂ O moderator.
Purpose:	Lattice studies
Size:	Tank 5' dia. x 7' high.
Fuel:	Any
Startup:	1953
Features:	Variable lattice pitch. Moderator temperature range 10 to 90°C. TTR type (SP) pile used as neutron source.
Owner/Operator:	USAEC/E.I. du Pont de Nemours and Co.

FIG. 9-5 PRESSURIZED SUBCRITICAL EXPERIMENT

Type:	Pressurized Exponential Tank
Purpose:	Lattice Studies
Size:	Tank 5 1/2' dia. x 12' high.
Startup:	1959
Features:	Max. pressure 300 psig. Max. temp. 215°C. Variable lattice pitch. Sb-Be neutron sources.
Owner/Operator:	USAEC/E.I. du Pont de Nemours and Co.

FIG. 9-6 RECEIVING BASIN FOR OFF-SITE FUELS

FIG. 9-7 HIGH LEVEL CAVES

FIG. 9-8 FUEL FABRICATION LABORATORY

FIG. 9-9 HIGH PRESSURE LOOP

FIG. 9-10 FLOWSHEET FOR COMPUTER STUDIES

10. BIBLIOGRAPHY

Listed below are references that should be helpful for further investigation of subjects covered in the text. The references are arranged according to the sections in the text.

2. Reactor Design

- 2.1 Hood, R. R. Heavy Water Moderated Power Reactors Progress Report November-December 1963. USAEC Report DP-885 (1964).

A conceptual design of a 1000-MWe liquid-D₂O-cooled reactor is given on pp. 16-17, and a detailed listing of the design conditions is given in the Appendix.

- 2.2 Wade, J. W. A Computer Program for Economic Studies of Heavy Water Power Reactors. USAEC Report DP-707 (1962).

This report describes the computer program that has been used for optimizing conceptual designs and for making preliminary economic evaluations.

- 2.3 Babcock, D. F., et al. An Evaluation of Heavy-Water-Moderated Power Reactors - A Status Report as of March 1963. USAEC Report DP-830 (1963).

A discussion is given of the reactor technology and the reactor designs for liquid-D₂O-cooled reactors in sizes of 300 and 500 MWe in Section 3.1, and a detailed listing of the design is given in Table A-1.

- 2.4 Babcock, D. F., et al. An Evaluation of Heavy-Water-Moderated Power Reactors - A Status Report as of March 1963. USAEC Report DP-830 (1963).

A discussion is given in Section 3.3 of the design optimization for the liquid-cooled reactor and for the effect that some of the design parameters have on the economics.

- 2.5 Wade, J. W. "Design Trends for D₂O Reactors". Trans. Am. Nuclear Soc. 5, 435 (1962).

This paper gives the results of some of the optimization studies that have been made on liquid-D₂O-cooled reactors, showing the effect of capital charge rates, natural uranium, and the use of uranium metal fuel.

- 2.6 Isakoff, L. and H. F. Measley. Heavy Water Moderated Power Reactors Progress Report October 1962. USAEC Report DP-795 (1963).

The results of optimization studies on a 300-MWe power reactor with liquid D₂O coolant and natural uranium oxide fuel are given on pp. 14-15 and in Figures 9, 10, 11, 12, and 13. The results of a similar study but with low capital charge rates are given in the Progress Report for November, 1962: DP-805, pp. 20-22.

- 2.7 Babcock, D. F., et al. An Evaluation of Heavy-Water-Moderated Power Reactors - A Status Report as of March 1963. USAEC Report DP-830 (1963).

In Section 2.1 of this report, an economic comparison is made of D₂O-moderated power reactors using liquid D₂O, boiling D₂O, organic, and H₂O fog coolants for 300- and 500-MWe plants. The selection of liquid D₂O coolant for future development is discussed in Sections 7.1-7.6. A detailed comparison of the conceptual designs and costs is given in Table A-1.

- 2.8 Hood, R. R. Heavy Water Moderated Power Reactors Progress Report November-December 1963. USAEC Report DP-885 (1964).

A comparison of liquid D₂O, boiling H₂O, and organic-cooled heavy-water-moderated power reactors at 1000 MWe is given on pp. 16-17 with a detailed listing of the design and cost estimates in the Appendix.

- 2.9 Hood, R. R. and L. Isakoff. Heavy Water Moderated Power Reactors Progress Report March 1962. USAEC Report DP-725 (1962).

Economic comparisons are given on pp. 25-26 and in Table V between (1) a liquid-D₂O-cooled reactor using enriched uranium oxide fuel, (2) a liquid-D₂O-cooled reactor using uranium metal fuel, and (3) a boiling-D₂O-cooled reactor with uranium oxide fuel.

- 2.10 Babcock, D. F., et al. An Evaluation of Heavy-Water-Moderated Power Reactors - A Status Report as of March 1963. USAEC Report DP-830 (1963).

Preliminary cost estimates are given in Section 2.3 for the use of D₂O reactors for process heat. The reactor technology associated with these reactor systems is given in Section 3.7.

- 2.11 Hood, R. R. and L. Isakoff. Heavy Water Moderated Power Reactors Progress Report March-April 1963. USAEC Report DP-845 (1963).

An economic evaluation is given on pp. 11-14 of the use of D₂O reactors with liquid D₂O, boiling H₂O, or organic coolants for use in desalination plants.

- 2.12 Large Reactor Study for Sea Water Distillation - Final Report. Bechtel Corporation, San Francisco (1963).

This report compares the various reactor types for use in saline water plants.

3. Size Considerations

- 3.1 Babcock, D. F., et al. An Evaluation of Heavy-Water-Moderated Power Reactors - A Status Report as of March 1963. USAEC Report DP-830 (1963).

The potential cost reduction due to increases in reactor size is discussed in Section 5.1. The cost for 300-MWe plants appropriate for prototype size is given in Section 2.1.

- 3.2 Hood, R. R. Heavy Water Moderated Power Reactors Progress Report November-December 1963. USAEC Report DP-885 (1964).

The cost reduction for a 1000-MWe plant relative to a 300-MWe plant plotted in Figure 2-1 was obtained from the cost estimates given in Table VII.

4. Use of H₂O Coolant

- 4.1 Babcock, D. F., et al. An Evaluation of Heavy-Water-Moderated Power Reactors - A Status Report as of March 1963. USAEC Report DP-830 (1963).

The design and cost estimates for a conceptual design using H₂O rather than D₂O coolant are given in Section 3.4.

5. Fuel Cycle Technology

- 5.1 Hood, R. R. Heavy Water Moderated Power Reactors Progress Report November-December 1963. USAEC Report DP-885 (1964).
- 5.2 Hood, R. R. Heavy Water Moderated Power Reactors Progress Report September-October 1963. USAEC Report DP-875 (1964).
- 5.3 Caskey, G. R., Jr., et al. "Failures of UO₂ Fuel Tubes by Internal Hydriding of Zircaloy-2 Sheaths", Symposium on Powder Packed Uranium Dioxide Fuel Elements, November 30-December 1, 1961. Section IV-E of CEND-153, Vol. II (1962).
- 5.4 Holmes, W. G. "Control of Hydrogen in the Fabrication of Tubular UO₂ Fuel Elements", Symposium on Powder Packed Uranium Dioxide Fuel Elements, November 30-December 1, 1961. Section IV-F of CEND-153, Vol. II (1962).
- 5.5 Hood, R. R. and L. Isakoff. Heavy Water Moderated Power Reactors Progress Report April 1962. USAEC Report DP-735 (1962).
- 5.6 Cole, G. R. "Influence of Thermal Rating and Bulk Density on Irradiation Performance of Fused UO₂ Tubes", Symposium on Powder Filled Uranium Dioxide Fuel Elements, November 5-6, 1963.

- 5.7 Hood, R. R. Heavy Water Moderated Power Reactors Progress Report January-February 1964. USAEC Report DP-895, p. 10 (to be issued).
- 5.8 Hood, R. R. and L. Isakoff. Heavy Water Moderated Power Reactor Progress Report April 1962. USAEC Report DP-735, p. 12 (1962).

6. Status of Physics Development

- 6.1 Crandall, J. L. Efficacy of Experimental Physics Studies on Heavy Water Lattices. USAEC Report DP-833 (1963).

A review is given of the requirements, methods, and accuracy of experimental studies performed on heavy water reactor lattices.

- 6.2 Brooks, W. L., et al. Present Status of Calculation and Experiment for D₂O Moderated Natural Uranium Lattices. USAEC Report UNC-5062 (1963).

Comparisons are made between experimental and calculated values of the reactivity for natural-uranium D₂O-moderated lattices.

- 6.3 Crandall, J. L. Status of the United States Effort in D₂O Reactor Physics. USAEC Report DP-787 (1962).

A summary is given of the current status and accomplishments of the U. S. program in D₂O reactor physics.

- 6.4 Graves, W. E., et al. "Experimental Bucklings and Void Effects in Heavy Water Lattices of Natural Uranium Oxide Rod Clusters". Nuclear Sci. and Eng. 16, No. 2, 186-95 (1963).

- 6.5 Baumann, N. P. PDP Measurements on Natural Uranium Fuel Tubes in D₂O. USAEC Report DP-640 (1961).

- 6.6 Hurley, T. J., et al. "Experimental Bucklings of Heavy Water Moderated Lattices of Natural Uranium Metal Rod Clusters". Nuclear Sci. and Eng. 12, 341-47 (1962).

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- 6.8 Soodak, H. and R. Sullivan. A Method for Calculating the Reactivity of D₂O-Moderated Natural Uranium Lattices with Clustered Rod Fuel Elements. USAEC Report NDA-2131-38 (1961).
- 6.9 St. Paul, R. G. and S. A. Rajé. NDC -- A FORTRAN Program for Unit Cell Nuclear Design Calculations for Rod-Clustered Pressure Tube Lattices. USAEC Report CVNA-146 (1962).
- 6.10 Rajé, S. A. KERNMAT -- A FORTRAN Code for the Heterogeneous Method of Reactor Calculations. USAEC Report CVNA-107 (1961).
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9. Savannah River Facilities for Reactor Development

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- 9.2 Dunklee, A. E. The Heavy Water System of the Process Development Pile. USAEC Report DP-567 (1961).
- 9.3 Towler, O. A., Jr. and J. W. Wade. "Exponential Measurements in Heavy Water Systems". Chem. Engr. Progr. Symposium Ser. 52, No. 19, 177-81 (1956).

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