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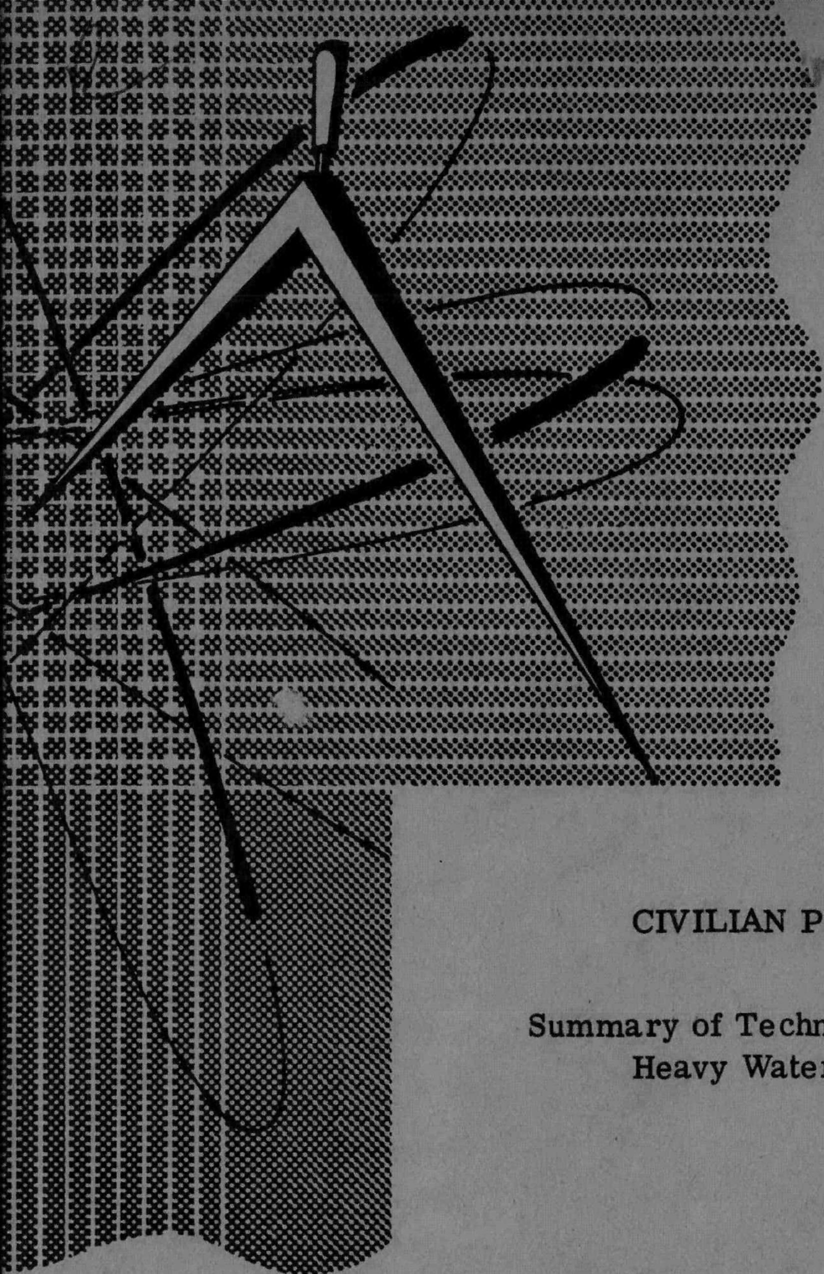

CIVILIAN POWER REACTOR PROGRAM

Part I
Summary of Technical and Economic Status as of 1960
Heavy Water-Moderated Power Reactors

Compiled by:

J.H. Hutton
S.A. Davis
C.C. Graves
J.G. Duffy

Reviewed by SROO

August 19, 1960

Work Performed for the
Savannah River Operations Office of the USAEC
under Contract AT(30-1)-2303(XIV)

NDA -
NUCLEAR DEVELOPMENT CORPORATION OF AMERICA
WHITE PLAINS, NEW YORK

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CONTENTS

TABLES

FIGURES

8. HEAVY WATER-MODERATED, NATURAL URANIUM-FUELED REACTORS

This report is a revision to Section 8 of TID-8516, Part I, Civilian Power Reactor Program. The reactor concept presented has been changed from a pressurized, pressure vessel, indirect cycle plant to a boiling, pressure tube, direct cycle plant. While the pressure tube concept has not, at the date of this writing, been demonstrated in an operating civilian power reactor it is, nevertheless, considered to be representative of current technology available in separate programs. For example, all prototypes or power reactors in the U.S. and Canadian programs, both those in construction and those being designed, are of the pressure tube type. Also, the PRTR, which is a typical pressure tube reactor, will operate in 1960 and the NPD-2 will operate in 1961. In addition, the Douglas Point plant (CANDU) includes a full-scale pressure tube reactor and is scheduled for construction completion by 1965.

The pressure tube reactor is, therefore, representative of D_2O-moderated power reactors and is now being used for the reference design in the ten-year program.

A. DESCRIPTION

In the few years that heavy water-moderated power reactors have been under investigation, virtually every combination of coolant and reactor core configuration has been studied, several reactor projects have progressed into construction, and improvements on these are being developed. All of the plants which are either under construction or scheduled for construction are cooled by pressurized D_2O. However, design studies (Reference SL-1565, SL-1776/NDA 2131-6, DP-480) by both du Pont and S&L-NDA, independently, indicate that the boiling D_2O-cooled, direct cycle plant holds the greatest promise for producing economic power in the near future.

These studies also indicate that heavy water-moderated natural uranium reactors must be large to be competitive, leading to the selection of the pressure tube design to avoid the pressure vessel size limitation. A logical direction for future growth from the boiling D_2O, direct cycle plant is to the use of H_2O fog coolant, possibly combined with nuclear superheat.

The cost data for the recommended concept presented herein is based on the boiling D_2O-cooled, pressure tube reactor plant studies conducted by du Pont, Sargent & Lundy and NDA since the Fall of 1958. The flow diagram and heat balance for the 300 MW_e plant are shown in Fig. 1; the reactor elevation is shown in Fig. 2; plant parameters are summarized in Table 1.

Cost data are presented for two plant sizes: 100 and 300 MW_e nominal net ratings with average discharge fuel burnups of 8500 MW-d/metric ton and 7500 MW-d/metric ton. The 300 MW_e plant data were obtained from SL-1815 with slight modifications to account for changes in site location. The 100 MW_e plant data were obtained by interpolating between the 300 MW_e plant and the 70 MW_e prototype presented in SL-1773. Both of the large plants are fueled with natural uranium.

Table 1 — Summary of Plant Characteristics – 110 and 325 MW_e Direct Cycle Plants

Description	D_2O Moderated 110.0 MW_e Gross	325.0 MW_e Gross
Heat balance		
Total reactor power, MW_t	365	1115
Gross turbine power, MW_e	110	340
Net plant power, MW_e	103	318.9
Net plant efficiency, %	28.2	28.6
Turbine cycle conditions		
Throttle temp, °F	510	510
Throttle pressure, psia	765	765
Total steam flow, lb/hr	1.51×10^6	4.61×10^6
Condenser back-pressure, in. Hg A	1.5	1.5
Final feedwater temp, °F	387	387
No. of feedwater heating stages	4	4
Reheat – temp, °F	—	—
Reheat – pressure, psia	—	—
Reactor description		
Reactor vessel		
ID, ft	16.8	20.6
Inside height, ft	16.3	22.5
Wall thickness, in. (cylindrical portion)	0.375	0.375
Material	Al	Al
Design – pressure, psig	15	15
Design – temp, °F	150	215
Type	Calandria	Calandria
Reactor core		
Active equivalent diameter, ft	12.8	18.6
Active height, ft	11.8	20.1
Active core volume, ft^3	1520	5450
Total uranium loading, kg U	22,230	85,700
Avg U^{235} content, % by weight	0.72	0.72
Structural material (pressure tubes)	Zr-2	Zr-2
Moderator to fuel ratio	14.9	13.9
Lattice arrangement	Triangular	Triangular
Total no. of lattice positions	173	369
Total no. of fueled positions	154	344
Reflector or blanket		
Material	D_2O	D_2O
Axial thickness, ft	2	1
Radial thickness, ft	2	1
Fuel elements (for each type)		
Fuel material	UO_2	UO_2
Fuel element geometry	Rods	Rods
Clad material	Zr-2	Zr-2
Fuel "meat" diameter, in.	0.500	0.500
Clad thickness, in.	0.025	0.025
Fuel-clad gap (cold), in.	0.005	0.005
Gap filler material	He	He
Fuel assemblies (for each type)		
Total no. (two per lattice position)	308	688
No. of elements (rods) per assembly	37	37
Cross sectional dimensions, in.	4.462 across hex. end points	4.462 across hex. end points
Lattice spacing, in.	11.1	11.1
End fitting materials	Zr-2	Zr-2

Table 1 — (Continued)

Description	D_2O Moderated 110.0 MW_e Gross	325.0 MW_e Gross
Reactor control		
Method of control	Rods	Rods
Absorber material	0.03 in. Cd (Al clad)	0.03 in. Cd (Al clad)
No. of control elements	19	25
Cross sectional dimensions, in.		
Effective length, ft	11.8	20.1
Type of drive	Motorized	Motorized
Calandria tubes		
Material	Al	Al
ID, in.	4.650	4.650
Wall thickness, in.	0.162	0.162
Pressure tubes		
Material	Zr-2	Zr-2
ID, in.	4.650	4.650
Wall thickness, in.	0.162	0.162
Coolant moderator insulation		
Material	Air	Air
Thickness, in.	0.375	0.375
No. of insulating gaps	1	1
Gap separators	Calandria and pressure tube annulus	Calandria and pressure tube annulus
Gap separator thickness	—	—
Performance data		
Reactor coolant outlet temp, °F	515	515
Reactor coolant inlet temp, °F	498	498
Primary system operating pressure, psig	795	795
Primary coolant flow, lb/hr	10.8×10^6	33×10^6
Avg core coolant velocity, ft/sec	6.37 at inlet	8.71 at inlet
Max fuel center temp, °F	4400	4500
Max cladding temp, °F	550	550
Burnout heat flux, Btu/hr-ft^2	Not available	1.025×10^6
Max core heat flux, Btu/hr-ft^2	3.11×10^5	3.18×10^5
Avg core heat flux, Btu/hr-ft^2	1.42×10^5	1.10×10^5
Avg core power density, kwt/ft^3	240	205
Peak to average power ratio	2.41	3.18
Avg specific power, kwt/kg U	16.4	13.0
Fuel management	Off power 4-zone radial shift axial repositioning	Off power 4-zone radial shift axial repositioning
Avg fuel burnup, MW-d/metric tons	6010	7500 and 8500
Peak to avg burnup ratio	Not available	Not available
Secondary sodium inlet temp, °F	—	—
Secondary sodium outlet temp, °F	—	—
Secondary sodium flow, lb/hr	—	—
Reactor coolant makeup rate, lb/day (D_2O)	30.2	57.3
Radial max to avg flux	1.59	1.94
Axial max to avg flux	1.38	1.49
Bundle max to avg flux	1.1	1.1
Max to avg thermal neutron flux	2.42	3.18
Power to coolant, MW_{th}	338	1037
Power to moderator and reflector	27	78
Containment		
Design criteria	Vapor containment	Vapor containment
Type	Steel shell	Steel shell
Primary loop coolant inventory, lb	190,000	361,308
Geometry	Cylindrical	Cylindrical
Dimensions, ft	114 $\phi \times$ 168 h	135 $\phi \times$ 190 h
Design pressure	~25 psia	31 psia
Material	Steel	Steel

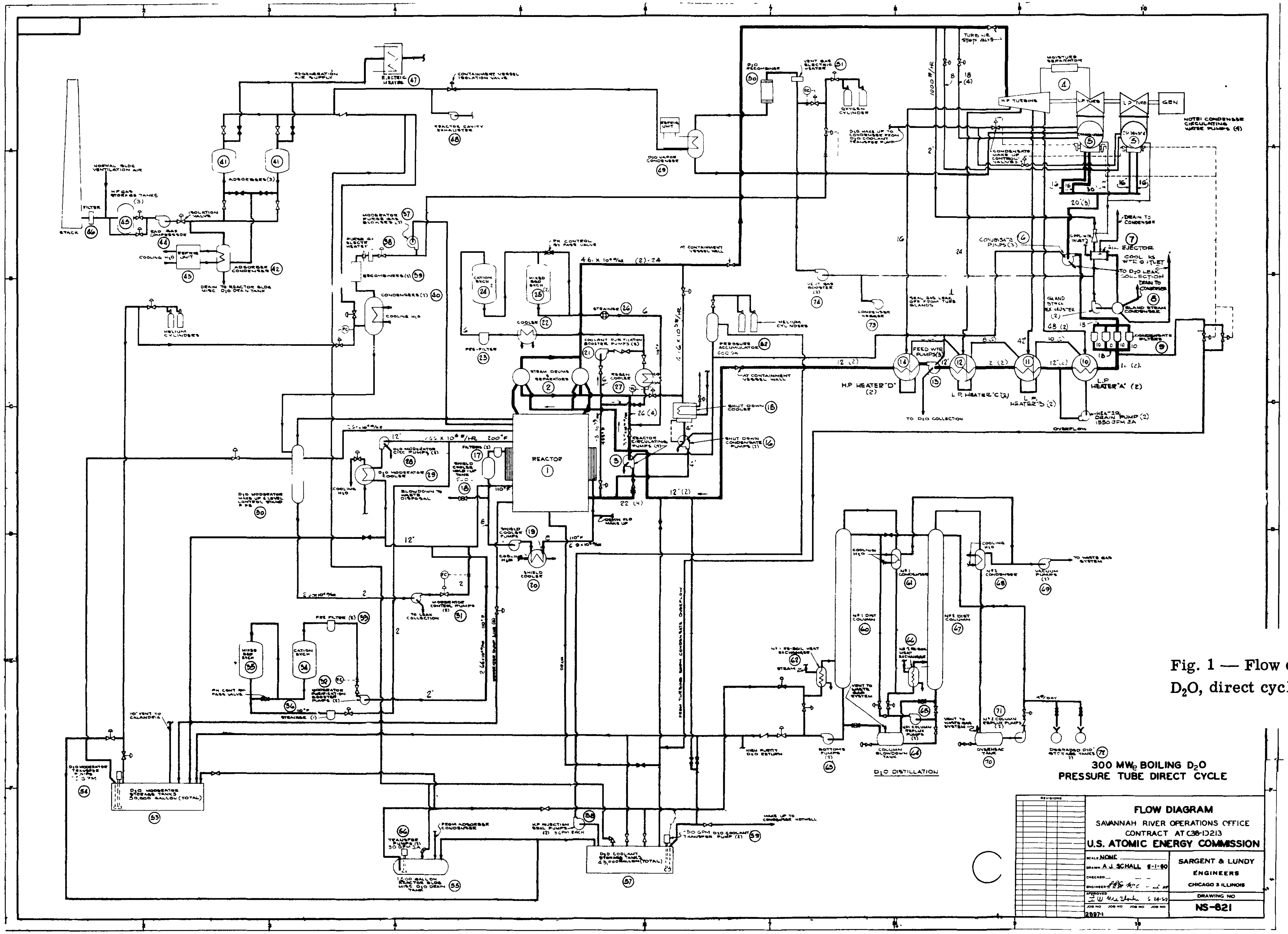

Fig. 1 — Flow diagram – 300 MW$_e$ boiling D_2O, direct cycle plant (preliminary)

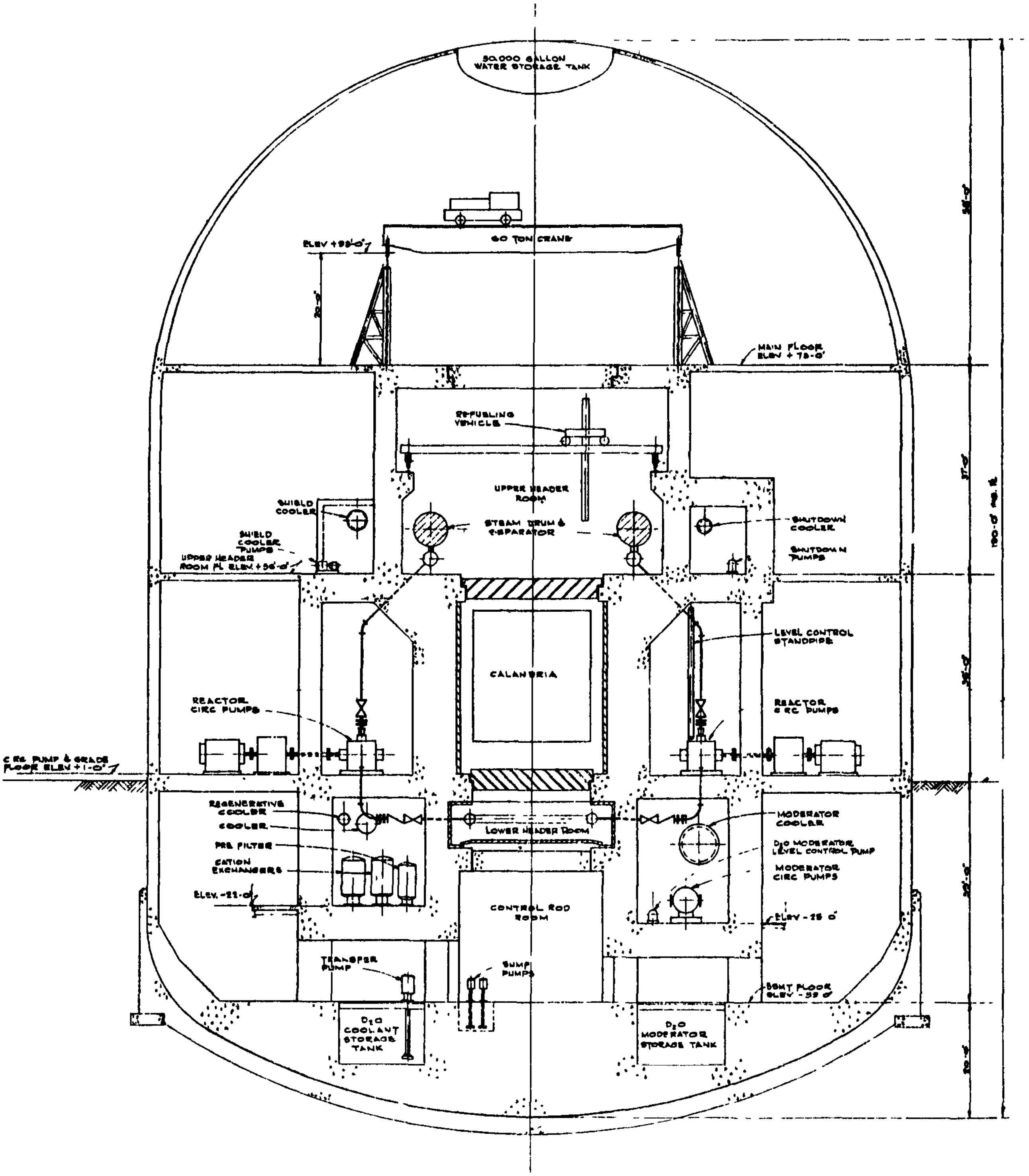

Fig. 2 — General cross section of reactor building – 300 MW_e boiling D_2O, pressure tube, direct cycle

B. TECHNICAL STATUS

The technical status of heavy water reactors is based on development programs being carried forward in the construction of the reactors mentioned in Section D plus general development projects at du Pont, Sargent & Lundy and NDA. The status of the main development efforts in these programs is as follows:

1. Physics – Reactivity Prediction

The methods of reactivity prediction currently available to the designer of natural uranium-fueled, heavy water reactors are semi-empirical in nature and are limited in their applicability. Some of the lattice designs for an oxide-fueled reactor are outside the range for which experimental data are available, and extrapolations to these designs may not be reliable. For example, the French and Swedish methods for prediction of reactivity agree closely for 19-rod clusters of 0.5 in. UO_2 fuel rods, but disagree by 1.5% k_{eff} for 37-rod clusters with lattice pitches of about 1 ft. The lattice configurations for which the existing semi-empirical methods can be used with confidence are compared in Table 2 with lattices of current interest.

Table 2 — Comparison of Present Lattice Designs with Valid Range of Reactivity Prediction Methods

	Fuel Area per Cluster, in.^{2}		Lattice Pitch, in.	
	Oxide	Metal	Oxide	Metal
Calculation Method				
Swedish	4.0-5.6	2.3-4.0	6.7-10.6	5.9-11.4
French	2.8-5.6	2.3-4.0	6.7-10.6	5.9-11.4
SRL-Canadian	1.4-6.5	0.9-9.3	7.5-11.8	7.5-11.8
Reactor Design				
S&L-NDA	8.8	—	11.1	—
Du Pont 1K-300	—	3.1	—	8.5

In the design of a natural uranium-fueled reactor, the expedient of increasing enrichment to overcome uncertainties in reactivity prediction is not available, but additional reactivity can be obtained for the cases considered by increasing the D_2O-to-uranium ratio. There is no doubt, from the reactivity standpoint, that a D_2O-moderated, natural uranium-fueled reactor can be built and operated. The question is rather one of specifying the optimum lattice and improving the predictions of attainable fuel exposure.

Current work to obtain more accurate reactivity data consists of the modification and operation of the Process Development Pile (PDP) at Savannah River for full scale lattice studies; design, construction and operation of the Pawling Lattice Test Rig (PLATR); analyses of various refueling schemes; and theoretical studies of reactivity prediction methods. Critical experiments with full scale lattices of uranium metal clusters have been run in the PDP; UO_2 experiments will start in mid-1960. PLATR went critical in March 1960 and is being used for UO_2-fueled lattice investigations. Data obtained from these facilities and from exponential experiments in the SE and PSE are expected to reduce the uncertainty in calculating initial reactivity to less than 1% k_{eff}. This uncertainty is equivalent to 5 to 10% in fuel burnup.

The current program will provide essentially all of the basic reactivity data which can be obtained conveniently without operating a reactor. Sufficient reactivity data from an experimental program should be available by the end of FY 1961 to specify optimum lattices for a prototype.

NDA studies of multizone refueling schemes indicate that a burnup of about 7500 MW-d/metric ton-U is probable with UO_2 in a 200 MW_e boiling reactor with off-power refueling. The Canadians predict 8000 MW-d/metric ton-U (+20%, -0%) with on-power refueling.

Predictions of long-term reactivity may be in error by as much as 30% in exposure. Improvements in this accuracy can be expected as the experimental and theoretical work continues. Some of the investigations in the HWCTR will be directed along these lines, and a critical facility for measuring the reactivity of exposed PRTR elements is being considered at Hanford. However, the final determination of burnup limits can be achieved only by operating a natural uranium-fueled prototype reactor.

2. Fuels and Materials

Two materials, uranium oxide and uranium metal, are being developed as alternative fuels for the heavy water-moderated power reactor. Uranium oxide has good dimensional stability under irradiation and is highly resistant to attack by D_2O. On the other hand, uranium metal is advantageous because of its greater nuclear reactivity.

Uranium oxide is known to be acceptable for use in power reactors. This fact has been demonstrated by the successful use of the material in the PWR reactor and by extensive in-pile testing. Because oxide fuel elements are now relatively expensive to fabricate, development attention is being concentrated on fabrication processes that have cost reduction potential.

It is not known yet whether metal elements can achieve the desired exposures under power reactor conditions without excessive fuel failures. Also, there is a theoretical indication that metal fuel cannot be used in boiling D_2O reactors because of reactor stability considerations. An extensive program to resolve these questions will be undertaken. Metallic fuel elements may show economic promise with more advanced coolants such as H_2O fog, steam, gas, or organics.

3. Heat Transfer and Fluid Flow

To attain the maximum potential of a reactor in which the coolant boils, more data are needed on the burnout heat flux and the pressure drop characteristics.

Although the problem of burnout is not considered to be of crucial importance for boiling reactors fueled with oxide rods, available data at conditions of interest are not adequate for optimum design of the fuel element. Conservative estimates of the burnout safety factor indicate that the anticipated operating heat flux is about 40% of the minimum burnout heat flux. The relationship between coolant flow and pressure drop is not known accurately for the pressures, geometries, and thermal conditions considered desirable for a boiling reactor.

An experimental program has been initiated at SRL and at the Columbia University Engineering Research Laboratories to determine the burnout heat flux and the pressure drop characteristics of various fuel element configurations. The experimental program will be supplemented by continuing efforts to correlate heat transfer data for boiling burnout. Pressures as high as 1500 psi and steam qualities as high as 20% or more will be included in the range of variables. This program will provide pressure drop data later in 1960, and it is expected that heat transfer limits for rod clusters will be fairly well defined by early 1961. The experiments and analyses will be followed by proof tests of fuel assemblies in the boiling D_2O loop of the HWCTR.

Current data indicate that it will be possible to raise the allowable maximum heat generation rate for UO_2 fuel elements above that used in the design studies. As new information is developed under the Canadian and U.S. programs, using UO_2 fuel, it will be factored into the D_2O-moderated reactor design.

4. Coolant Chemistry

The requirements imposed on the purity and handling of D_2O as a reactor coolant are essentially the same as for H_2O. This technology is well established in boiling reactors such as EBWR, VBWR, and Dresden. Current work includes the specification of coolant conditions required for use of less expensive process system materials.

5. Components and Auxiliary Systems

Pressure Tubes. The zirconium-base alloys are currently considered to be the best available materials for pressure tubes in a natural uranium power reactor. No other commercially available metal that has adequate mechanical properties and corrosion resistance is sufficiently transparent to neutrons to be attractive for this purpose. Because of their high replacement cost, the pressure tubes must be capable of trouble-free service for many years. The limited irradiation data obtained thus far engender confidence that the service requirements can be met. However, zirconium alloys are relatively untried in reactor structural applications, and the effects of prolonged irradiation on their mechanical properties are not well known. As a consequence, opinions differ with respect to safe design stresses, especially for highly cold-worked material. The only way to resolve this question is to obtain in-pile data for large numbers of pressure tubes. Such data will be obtained from the PRTR, NPD-2, and CVTR, all of which will employ pressure tubes of zirconium alloy.

Most of the development work to date on Zircaloy pressure tubes has been conducted in support of the construction programs for the PRTR, CVTR, and NPD-2. Emphasis is being placed at present on inspection and evaluation of tubes which have been delivered for these reactors. Experience thus far indicates that pressure tube fabrication will not pose major problems. Of the 97 tubes delivered for the PRTR, only a few had minor defects and even these will be installed in the reactor and observed closely for incipient failures. The fabrication yield of PRTR tubes was such that the cost of the finished tubes was about \$60/lb of zirconium. For large orders of tubes, fabrication costs as low as \$25/lb are quoted. AECL has received about 20 tubes for the NPD-2, and evaluation results on these tubes will soon be forthcoming.

Irradiation data on pressure tubes are being obtained at Hanfora and at Chalk River. At Hanford, long tubes of Zircaloy-2 (2.1 in. ID) are being irradiated in test loops at a temperature of 430°F and a pressure of 900 to 1500 psi. One of these tubes was recently sectioned for examination after irradiation for about two years. Except for one section of the tube which had been exposed inadvertently to conditions that are extraneous to the power reactor program, the results of the examinations to date are reported to be generally satisfactory. A section that had been irradiated at the edge of the peak flux area exhibited no recrystallization or inclusions. There was no obvious change in tube dimensions, and no evidence of localized corrosion. Further studies of this tube are under way. At Chalk River, a 5-in. diameter Zircaloy-2 pressure tube has been in service in the NRX reactor for three years at 1800 psi and 520°F. No abnormalities have been detected in periodic visual inspections of the tube. It is understood that it will be removed for destructive evaluations later this year.

The direct evaluations described above are being supplemented by experimental studies at the various sites. General Electric is now beginning to obtain in-pile creep data on Zircaloy specimens at Hanford, and is initiating a similar program at KAPL. The Canadians are conducting 10,000 hr creep tests on unirradiated Zircaloy at relatively high stresses. Data from the latter tests will be available in late 1960, and will form the basis for specifying the design stress for the CANDU reactor. The immediate Canadian program includes burst tests of intentionally defected pressure tubes in a mechanical mockup of the NPD-2 lattice. These tests are pointed toward an evaluation of the consequences of an in-pile failure of a tube. Westinghouse and Nuclear Materials and Equipment Corporation are conducting out-of-pile test work on Zircaloy in connection with design development of the CVTR and FWCNG reactors, respectively. In cooperation with AECL, du Pont is measuring the stress relaxation of Zircaloy specimens during irradiation in the NRX reactor. In addition, two Zircaloy-4 pressure tubes are being procured for the isolated coolant loops of the HWCTR. Zircaloy-4, which contains no nickel, offers the potential advantage of lower hydrogen absorption in a reactor and may prove to be a better alloy for pressure tubes than Zircaloy-2.

Joints and Closures. Strong, leaktight connections are required in a pressure tube reactor to join the Zircaloy pressure tubes to the external piping of the reactor. These connections are difficult to accomplish because of the wide difference between the coefficients of thermal expansion of Zircaloy and stainless steel, and because the two materials cannot be joined by direct fusion welding. Both mechanical and metallurgically bonded joints are being developed for this application.

Favorable test results have been obtained in other reactor programs with conventional mechanical joints of several designs; the two Zircaloy-tubed reactors which have advanced beyond the study stage will utilize mechanical or rolled joints. The first performance data will be obtained during operation of the PRTR and the NPD-2. The joints for the PRTR are flanged connections in which Flexitallic gaskets are used as seals. In the NPD-2, the joint is made by rolling the Zircaloy into a series of grooves in an overlying stainless steel tube. Both of these joints have performed well under simulated service conditions, but recent results indicate that Zircaloy corrosion by stagnant water may be a problem with the rolled joint. A test program has also been initiated by Westinghouse on the mechanical joint that is contemplated for the CVTR. In this joint, the seal between Zircaloy and stainless steel is an adaptation of a conventional Marman Conoseal joint.

Metallurgically bonded joints between Zircaloy and stainless steel are attractive because their compactness permits closer lattice spacings and makes it possible to reduce the quantity of Zircaloy adjacent to the reactor core. Rapid progress has been made in recent months at Nuclear Metals, Inc. in the development of metallurgically bonded joints, and specimens of tubular joints of practical size are being evaluated. In a burst test, one specimen of a bonded joint (1.9 in. OD × 0.2 in. wall) withstood an internal pressure of 16,500 psi at low temperature without failure of the joint. Two other samples have been cycled to 1000 psi and 500°F about 100 times without measurable leakage of water. The corrosion resistance of the bonded joint appears to be good. The greatest uncertainty is possible hydrogen embrittlement of the Zircaloy as a result of nickel diffusion from the stainless steel.

The program on metallurgically bonded joints includes irradiation tests as well as more extensive burst tests, corrosion tests, and thermal cycling tests. Irradiation tests under power reactor conditions are planned, and earlier irradiations at lower temperatures in a Savannah River reactor are being considered.

D_2O Leakage. Heavy water is such an expensive commodity that its unrecoverable loss from a reactor plant is an item of great concern, particularly since no operating experience has been gained at conditions of interest. The economic import of D_2O losses is shown in Fig. 3, which relates the loss rate to power costs for reactors that are cooled by boiling D_2O. The losses are also objectionable because of attendant tritium hazards.

Quantitative measurements of water leakage from individual components for a reactor plant have been made. The principal objective of this program was to improve the reliability of estimates of overall loss in a full scale reactor. A secondary objective was to secure data which would facilitate design of D_2O handling equipment, recovery facilities, and ventilation systems. Concurrently, a similar investigation of leakage from selected components of the HWCTR was conducted. AECL and GE have investigated component leakage. These programs have provided data on leakage rates through static joints and closures of conventional design, valve stems, pump seals, turbine seals, and tubing fittings. S&L and du Pont have been conducting tests on turbine

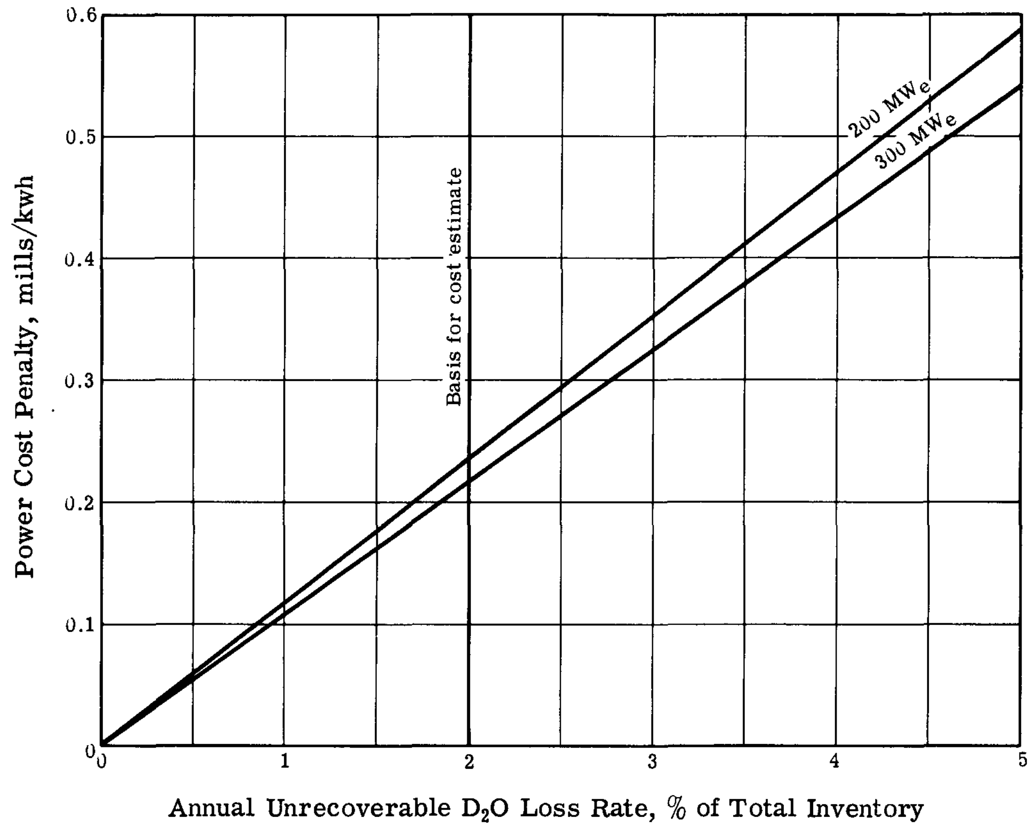

Fig. 3 — Effect of D_2O loss rate on power cost for 300 MW_e oxide-fueled, boiling D_2O, direct cycle plants

seals, pump seals, and valves. The data accumulated to date shows encouraging results and indicates that control of leakage will not be a major problem.

The measurements of leakage from individual components are eliminating much of the uncertainty with respect to unrecoverable D_2O losses. However, the results provide no clue to the losses that will result from operating errors and from equipment malfunctions, abnormalities, and failures. HWCTR, CVTR, and NPD-2 designs incorporate means that insure tight control over D_2O leakage. Operational experience with these systems will provide further information on the controllable D_2O loss rate. Data obtained to date indicates that the D_2O loss rate assumed in the economic studies should be reduced.

Fueling. The economics of a power reactor are affected significantly by the average fuel burnup. A natural uranium reactor must make use of a fuel shuffling program in order to take full advantage of the limited reactivity available for fuel burnup. Maximum burnup can be obtained by continuous refueling, which must be performed during reactor operation. However, on-power refueling adds to the cost and complexity of a power reactor plant. It is currently estimated that an average exposure of 8500 MW-d/metric ton-U can be achieved with natural uranium in a 300 MW_e, oxide-fueled, boiling reactor through the use of a four-zone, outward radial shift, off-power refueling scheme (see Section 7.1, NDA 2153-3). An additional ground rule required that the maximum fuel burnup should not exceed 8500 MW-d/metric ton-U. Therefore cost data are presented for both 8500 and 7500 MW-d/metric ton-U. The Canadians expect to achieve even higher exposures as well as improved innage through use of a countercurrent, on-power refueling plan that is to be demonstrated in the NPD-2. A prototype of the refueling machine for the NPD-2 is under test.

Further studies of alternative fuel scheduling schemes will be made to determine which are most economical. In addition, the Canadian development of on-power refueling will be followed closely.

6. Safety

Heavy water-moderated-and-cooled reactors are relatively slow in responding to disturbances and are easily controlled. In general, however, these reactors have small but positive coolant void coefficients, and each type of design must be examined to determine whether this positive coefficient introduces any control problems. For the boiling D_2O reactors, the existence of this characteristic has raised questions as to (1) whether positive reactivity feedback through the void coefficient can lead to an uncontrollable power excursion, and (2) whether local perturbations in flow will give rise to local changes in steam quality which, through the positive void coefficient, will lead to local power increases and burnout.

Detailed studies conducted at NDA during the past year have shown that such problems are of minor consequence for an oxide-fueled reactor. The following conclusions for an oxide-fueled reactor may be drawn from these studies:

1. The net power coefficient of reactivity is negative. The negative power coefficient resulting from an increase in temperature of the oxide fuel is approximately five times that of the positive power coefficient resulting from coolant steam void formation at design power conditions.
2. The reactor is self-stabilizing. After a step insertion of reactivity, a new steady power level is attained without need for control rod motion. Power changes are slow and overshoots are small as a consequence of the large thermal time lag in the oxide fuel (high heat capacity, low thermal conductivity). The maximum fuel surface heat fluxes for the transients that were investigated were well below the boiling burnout limits.
3. The above-stated conclusions are qualitatively the same even when the estimated void or temperature coefficient is varied by a factor of two in the unfavorable direction.
4. The maximum local changes in power resulting from the interaction of coolant vapor formation with the positive void coefficient are less than 10%. Under these conditions there is no possibility of a local power perturbation leading to boiling burnout.

The metal-fueled, boiling D_2O reactor has not been fully investigated, but preliminary analyses indicate that in such a reactor the positive void coefficient will not be overridden by the negative fuel temperature coefficient.

Further refinement and confirmation of calculation techniques are now being obtained. Void coefficients for rod clusters and nested tubes, as well as moderator temperature coefficients, will be experimentally determined.

The data on these coefficients will be obtained along with the constants for cold, clean lattices. Ultimate confirmation of the effects on reactor stability will be obtained when a prototype is tested.

C. OPERATING EXPERIENCE

Operating experience with D_2O-moderated reactors was summarized in last year's report, TID-8516, Part I.

Considerable experience can be drawn from the operating experience of other plants but the proof of a heavy water-moderated, natural uranium-fueled power reactor will come only from the operation of a prototype plant. The main areas in which further data are required prior to constructing a prototype plant are reactivity and Zircaloy in-pile performance.

D. PLANTS UNDER CONSTRUCTION

Six heavy water-moderated reactor projects are under way, or being considered, in North America, four in the U.S., and two in Canada. Detailed information on these reactors may be found in Table 7.1, Part III, of this report. The contribution of each reactor to the overall development program is discussed below.

1. Plutonium Recycle Test Reactor (PRTR)

The PRTR will be used to determine the practicality and economics of the plutonium recycle concept. It is fueled with Zircaloy clad natural UO_2 fuel with Pu-Al spikes. The reactor core is the tube type with a lattice of 19-rod clusters of 0.5 in. fuel rods clad with Zr-2, in Zr-2 pressure tubes. The coolant is pressurized D_2O. The moderator is contained in a calandria and is insulated from the coolant by a gas space.

The major contribution to the Zircaloy pressure tube fabrication technology was made in the development of PRTR. Also, swaged UO_2 fuel elements are being produced for this reactor and should result in a substantial fuel fabrication cost reduction.

The PRTR will be the first reactor to operate with conditions of temperature, pressure, and flow comparable to those found in a power reactor and will contribute to the D_2O handling technology. The plant uses mechanical pump shaft seals and Flexitallic gaskets between conventional flanges for the Zr-2-to-stainless steel pressure tube joints. The performance of these conventional components is quite important in determining and controlling D_2O losses and the measures to be taken in keeping them under control.

The cost of fabricating plutonium fuel elements is also of considerable interest because the recycling of plutonium through a reactor with natural uranium fuel would provide the increase in reactivity needed for high burnup or power flattening. Since the plutonium production rate is higher in a D_2O-moderated reactor than in most other types of thermal reactors, a greater advantage will be gained by the use of Pu recycle in this type of reactor.

2. Heavy Water Components Test Reactor (HWCTR)

The HWCTR is being built primarily to conduct fuel element irradiations on representative full length elements and to evaluate power reactor components. It is a pressurized, D_2O-moderated-and-cooled, pressure vessel type reactor. Twelve positions for natural uranium fuel elements up to 10 ft long are driven by a peripheral ring of enriched fuel elements. The reactor operates at power reactor conditions and has two separate loops; one for operation under boiling conditions and the other for operation at 500 psi higher than base reactor conditions (i.e., 1500 psi). The boiling loop is convertible, with some modifications, for use with high pressure gas coolants.

Virtually any type of fuel can be irradiated in the HWCTR. It is possible to run three fuel elements in parallel in one loop to determine parallel flow stability. Also, an attempt will be made to calibrate the core to obtain the effect of burnup on reactivity. Since the reactivity is provided by enriched fuel, it is possible to irradiate natural uranium fuel elements to any desired burnup.

3. Carolinas-Virginia Tube Reactor (CVTR)

The CVTR is a 17 MW_e, pressurized D_2O-cooled, pressure tube type reactor fueled with slightly enriched UO_2. It is a prototype for a larger, natural uranium-fueled reactor plant. The site has been selected and a construction permit has been issued and site preparation is under way.

A unique feature of the CVTR is the U-tube construction of the Zircaloy pressure tubes, making a two-pass reactor core. The coolant is insulated from the moderator by an internal stagnant D_2O space provided by four thin Zircaloy shrouds. All U-tube joints are special adaptations of a Marman "Conoseal."

4. Nuclear Power Demonstration Reactor (NPD-2)

The NPD-2 will be the first natural uranium-fueled, heavy water-moderated power reactor to operate in the Western Hemisphere. The core consists of a horizontal lattice of 19-rod UO_2 fuel elements in Zircaloy pressure tubes. On-power refueling will be used to improve fuel burnup and provide shim control of the reactor.

Operation of the NPD-2, scheduled for 1961, will provide the first proof of the reactivity characteristics of the reactor type. Long term reactivity of natural uranium is of great interest to the program and can be determined accurately only by operating a natural uranium-fueled reactor.

5. Florida West Coast Nuclear Group Gas-Cooled Reactor (FWCNG)

The FWCNG power plant utilizes an advanced CO_2-cooled, heavy water-moderated, pressure tube reactor. Both prototype and 300 MW_e plants are being investigated. Research and development leading to the construction of a 50 MW_e prototype is being conducted by General Nuclear Engineering Corporation and American Electric Power Service Corporation under an agreement with the East Central Nuclear Group.

The reactor provides 1050°F, 500 psi gas to a steam generator, thus producing 950°, 1450 psia steam to the turbine. In the initial concept a reheat cycle was used which had a net efficiency of 34.6% for the 300 MW_e plant. Recent optimization studies favor a nonreheat cycle. Conceptual design will be revised accordingly.

Plant refueling is accomplished while the reactor is in operation, the only U.S. project considering this concept. The high temperature coolant places severe performance requirements on the fuel cladding for operation on natural uranium. Beryllium alloy fuel cladding is being developed for the full scale plant and the prototype program has been reoriented to incorporate this type of cladding rather than stainless steel as originally planned.

6. CANDU Reactor

The CANDU is the only full scale power station currently planned for construction. It is a 200 MW_e plant with design conditions similar to the NPD-2. The site at Douglas Point has been selected and the reactor is scheduled for completion in 1965.

E. ECONOMICS

The cost estimates presented below represent the current status of heavy water-moderated power reactors. A series of evaluation studies were conducted by du Pont, S&L, and NDA. As a result of these studies it was indicated that the boiling D_2O-cooled, cold moderator, pressure tube, direct cycle plant was the most promising development for economic potential in the immediate future. A recent study, SL-1776/NDA 2131-6, showed that the boiling D_2O, pressure tube, direct cycle plant hadonly a small economic advantage (0.11 mill/kwh) over the identical indirect cycle plant. The selection of the direct cycle case was influenced by the inherent advantage of going to fog or steam coolant as a long range development. To compare U.S. and Canadian reactors on the same basis a cost analysis was made comparing a 200 MW_e U.S. plant design with CANDU, which is also a 200 MW_e plant. Using Canadian financing, the power generation costs differ only by 0.14 mill/kwh.

The cost data for the 300 MW_e plant was based on that given in reference SL-1815. Slight modifications to SL-1815 were required due to a change in site location and economic ground rules. The 100 MW_e plant data was obtained by interpolating between the 300 MW_e plant and the 70 MW_e prototype presented in Reference SL-1773.

The capital cost breakdown in accordance with the AEC system of accounts is given in Table 7.4 of the Part III report, NDA 2153-3. Capital and operating costs are summarized in Table 3.

Table 3 — Summary of Current Cost Estimates for Boiling D_2O, Pressure Tube, Direct Cycle, Natural UO_2-Fueled, Reactor Power Plants

	Nominal 325 Gross MW_e Plant (8500 MW-d/tonne Burnup) 2235×10^6 kwh/yr at 0.8 Operating Factor			Nominal 325 Gross MW_e Plant (7500 MW-d/tonne Burnup) 2235×10^6 kwh/yr at 0.8 Operating Factor			Nominal 110 Gross MW_e Plant (6010 MW-d/tonne Burnup) 722×10^6 kwh/yr at 0.8 Operating Factor		
	Investment $\$/10^6$	Annual Cost $\$10^6/yr$	Power Cost mills/kwh	Investment $\$/10^6$	Annual Cost $\$10^6/yr$	Power Cost mills/kwh	Investment $\$/10^6$	Annual Cost $\$10^6/yr$	Power Cost mills/kwh
Investment									
Plant investment	74.970	10.496	4.696	74.970	10.496	4.696	39.652	5.551	7.688
D_2O investment	22.839	2.855	1.277	22.839	2.855	1.277	12.164	1.520	2.105
Total investment	97.809	13.351	5.973	97.809	13.351	5.973	51 816	7.071	9.793
Operating									
Fuel costs		3.372	1.509		3.902	1.746		1.968	2.726
Heavy water makeup		0.468	0.209		0.468	0.209		0 247	0.342
Operating and maintenance payroll		0.751	0.336		0.751	0.336		0.571	0.791
Operating supplies and maintenance materials		0.546	0.244		0.546	0.244		0.380	0.526
Insurance		0 293	0.131		0.293	0.131		0.268	0.369
Working capital		0.456	0.204		0.454	0.203		0.140	0.194
Total operating costs		5.886	2.633		6.414	2.869		3.574	4.948
Total capital and operating costs		19.237	8.606		19.765	8.842		10.645	14.741