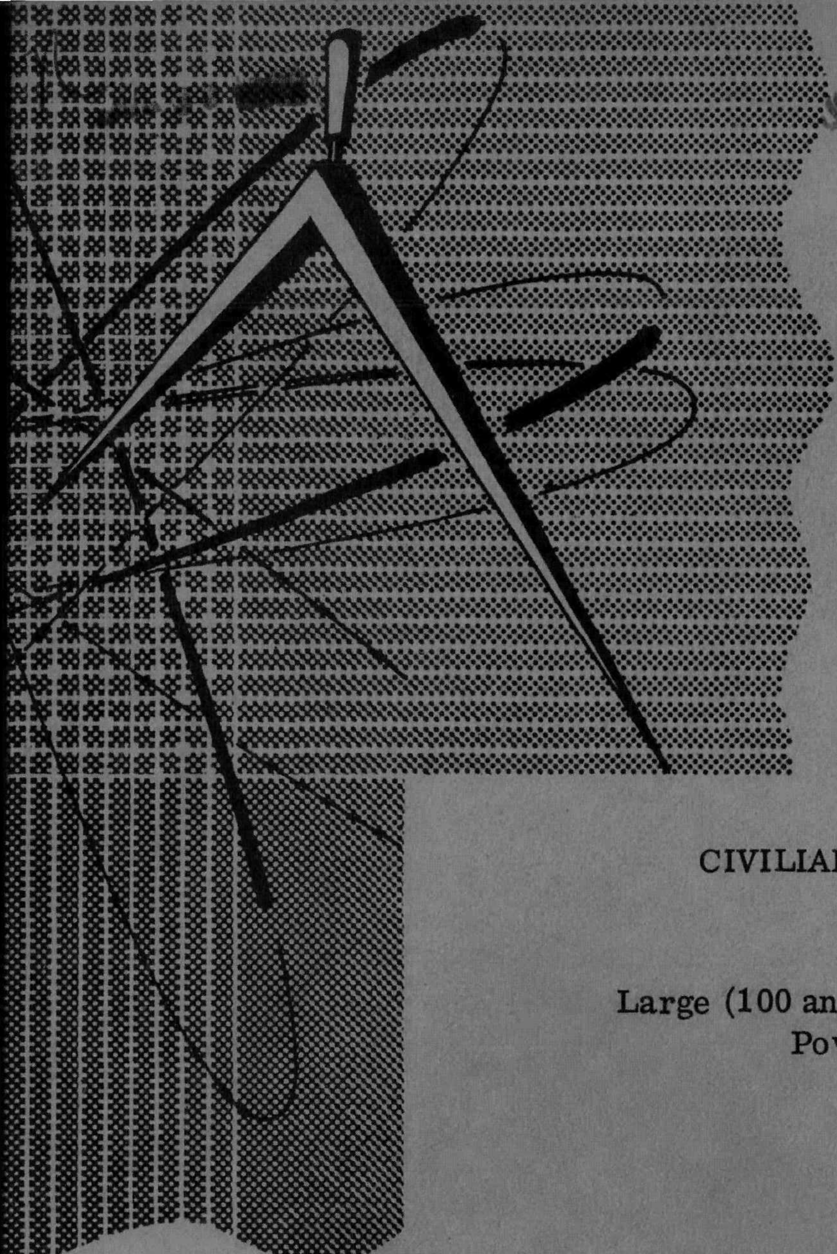

NDA 2153-3

CIVILIAN POWER REACTOR PROGRAM

Part III
Status Report on
Large (100 and 300 MW_e) Heavy Water-Moderated Power Reactors – As of 1960

Report 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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ABSTRACT

An evaluation of 300 and 100 MW_e power plants has been conducted using ground rules prescribed by the USAEC for this study. Costs corresponding to two average discharged fuel burnups are: 8.6 mills/kwh (8500 MW-d/metric ton) and 8.8 mills/kwh (7500 MW-d/metric ton) for the 300 MW_e plant. Costs for the 100 MW_e plant are 14.7 mills/kwh for an average discharged fuel burnup of 6010 MW-d/metric ton. Estimates of future potential indicate that the 300 MW_e (8500 MW-d/metric ton) plant could produce power for 7.3 mills/kwh in a second generation, full scale plant of the same type. A further reduction to 6.4 mills/kwh should be possible as the result of the recommended ten-year development program.

The current development program is adequate for providing the data needed to design and construct a prototype reactor. However, there is no natural uranium-fueled prototype and no prototype of the chosen reference design scheduled in the U.S.

Current technology is sufficiently developed to initiate the design and construction of a pressure tube, boiling D_2O-cooled, natural UO_2-fueled reactor prototype plant in the immediate future. This plant would demonstrate the main features of a full scale plant and, in addition, would provide design data which could only be obtained by operation of a natural uranium-fueled reactor.

ABBREVIATIONS

ORGANIZATIONS

AEC	The U.S. Atomic Energy Commission
AECL	Atomic Energy of Canada, Limited
CISE	Centro Informazioni Studi Esperienze
ECNG	East Central Nuclear Group
FWCNG	Florida West Coast Nuclear Group
GE	General Electric Company
GNEC	General Nuclear Engineering Corporation
HAPO	Hanford Atomic Products Operation
KAPL	Knolls Atomic Power Laboratory
NDA	Nuclear Development Corporation of America
NMI	Nuclear Metals, Incorporated
S&L	Sargent & Lundy, Engineers
SRL	Savannah River Laboratory
SRP	Savannah River Plant

REACTORS

CANDU	A 200 MW_e, heavy water, natural uranium-fueled power reactor presently scheduled to be built in Ontario, Canada by AECL. Startup is scheduled for early 1965.
CVTR	Carolinas-Virginia Tube Reactor, a 17 MW_e, heavy water, power demonstration reactor to be built near Columbia, South Carolina. Startup is scheduled for late 1962.
EBWR	Experimental Boiling Water Reactor, a 5 MW_e, light water, power reactor experiment now in operation at the Argonne National Laboratory.
ETR	Engineering Test Reactor at the National Reactor Testing Station.
HWCTR	Heavy Water Components Test Reactor under construction at SRP. Startup is scheduled for the third quarter of 1961.
MTR	Materials Testing Reactor at the National Reactor Testing Station.
NPD-2	A 20 MW_e, heavy water, power demonstration reactor under construction by AECL in Ontario, Canada. Startup is scheduled for early 1961.
NRU	A heavy water test reactor at the Chalk River Laboratory of AECL.
NRX	A heavy water test reactor at the Chalk River Laboratory of AECL.
PCTR	Physical Constants Test Reactor, a critical assembly at HAPO.
PDP	Process Development Pile, a full scale critical facility at SRL.
PLATR	Pawling Lattice Test Rig, a critical assembly installed in the Pawling Research Reactor.
PRR	Pawling Research Reactor, a heavy water research reactor at the Pawling (N.Y.) Laboratories of NDA.
PRTR	Plutonium Recycle Test Reactor, a heavy water test reactor at HAPO.
PSE	A pressurized heavy water exponential facility at SRL.

PWR Pressurized Water Reactor, a light water power demonstration reactor built by the Westinghouse Electric Corporation for the AEC-Duquesne Light Company at Shippingport, Pa.

SE A heavy water exponential facility at SRL.

VBWR Vallecitos Boiling Water Reactor, a boiling light water power demonstration reactor built by GE at their Vallecitos (Cal.) Laboratory.

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TABLES

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1. INTRODUCTION

This is one of three reports being submitted in response to a request from the Savannah River Operations Office of the U.S. Atomic Energy Commission for a review and up-dating of the three parts of the Civilian Power Reactor Program. This report is a revision of TID-8518, Part III, Technical Status of Heavy Water-Moderated Reactors, in accordance with the AEC 1960 Format. This report, which covers large, natural uranium plants, was compiled by NDA and approved by the Savannah River Operations Office.

Part I – Summary of Technical and Economic Status as of 1960, has been revised as NDA 2153-1; Part II – Economic Potential and Development Program, has been revised as NDA 2153-2. The work was conducted under AEC Contract AT(30-1)-2303(XIV).

In preparing this report, information was gathered from a large number of reports on the various projects in the program. In particular, DP-480, a status report on Heavy Water-Moderated Power Reactors, prepared jointly by E.I. du Pont de Nemours & Co., Sargent & Lundy, Engineers, and Nuclear Development Corporation of America, was used extensively. Also included is the heavy water power reactor technology being developed by other organizations, notably Atomic Energy of Canada, Ltd. (NPD-2 and CANDU); Westinghouse Electric Corp. (Carolinas-Virginia Tube Reactor); General Electric Co., Hanford Atomic Products Operation (Plutonium Recycle Test Reactor); American Electric Power Service Corporation and the General Nuclear Engineering Corporation (gas-cooled reactor for the East Central Nuclear Group and Florida West Coast Nuclear Group); and the du Pont Company (Heavy Water Components Test Reactor).

2. SUMMARY

The reactor concept which was presented as current technology for heavy water-moderated reactors in the 1959 program report was a pressure vessel, pressurized D_2O-cooled reactor in an indirect cycle plant. However, it has been shown by S&L-NDA and du Pont studies[7,43,46] that a pressure tube, boiling D_2O-cooled reactor in either a direct or indirect cycle plant has greater promise of producing economic electrical power. The boiling D_2O, direct cycle, pressure tube reactor has been chosen to be presented in this program report as a concept that would be technologically available in the immediate future.

While the boiling D_2O-cooled, 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, heavy water-moderated reactors under construction in the U.S. and Canada include a demonstration plant (NPD-2), two test reactors (HWCTR and PRTR), a prototype reactor (CVTR), and a 200 MW_e power station (CANDU). A conceptual design is in progress for the FWCNG gas-cooled, D_2O-moderated reactor. Design and development of boiling D_2O-cooled reactors is being carried out cooperatively by the du Pont Company, Sargent & Lundy, and NDA; the du Pont and S&L-NDA studies indicate that a pressure tube reactor cooled by boiling D_2O offers the most promise of eventually producing competitive electric power.

Of the various D_2O-cooled-and-moderated reactor systems that have been studied to date, it is estimated that a boiling reactor of this type fueled with natural uranium could produce power for 8.6 mills/kwh in a 300 MW_e plant and for 14.7 mills/kwh in a 100 MW_e plant.

When cost bases other than those currently formulated by the AEC are used, significant changes in these figures result. For example, the power cost for the 300 MW_e plant would decrease 0.9 mill/kwh if the heavy water could be leased at the same rate (4%/year) as specified for the lease of enriched uranium. The power cost would increase about 1 mill/kwh if liquid cooling were substituted for boiling.

The method of financing has a large effect on power costs. In the United States there are three bases of financing power stations: private utility, public utility, and REA. In the analysis of Canadian power costs (see Section 7.7, Table 7.10), it was observed that the capital charges employed are typical of publicly-owned utility systems. For these systems, the low fuel costs obtainable with D_2O-moderated reactors are particularly attractive since there is little power cost penalty from the higher capital investment required. For example, a 200 MW_e D_2O-moderated reactor plant would produce power for 11.3 mills/kwh in a privately-owned utility system employing 14% capital charges; the same plant would produce power for 5.8 mills/kwh in a publicly-owned utility system such as Ontario Hydro in Canada.

Second generation plants of the boiling-D_2O type should produce power at 7.3 mills/kwh in the 300 MW_e size. Improvements resulting from the development program, exclusive of concepts such as plutonium recycle, throium-uranium cycle, and vastly improved materials, would make it possible to design a 300 MW_e plant in 1970 which would produce power at 6.4 mills/kwh.

The current research and development program has included significant work in the following areas:

1. Fuel Elements

A very important problem in the reduction of power cost is the fuel element. In order to attain competitive power costs from a D_2O-moderated reactor fueled with natural uranium, the fuel cycle cost must be made lower than the cost of either fossil fuel or the enriched uranium fuel used in other power reactors. The economic data presented herein are based on the use of UO_2 pelletized fuel in Zr-2 cladding. In addition, two possible routes to lower cost fuel are being investigated:

> The swaging process for production of fuel tubes and rods of uranium oxide is being developed as a potential means of decreasing the fabrication costs of oxide elements. Uranium oxide is of interest because of its excellent resistance to radiation damage and to corrosion by water, but existing fabrication processes for oxide elements are costly. Low fabrication costs for natural uranium fuel are an economic necessity because the attainable burnup is less than that generally predicted for slightly enriched fuels.
>
> Efforts are being made to develop acceptable fuel elements of uranium metal. This material offers higher nuclear reactivity; however, its ability to withstand the desired burnup under power reactor conditions has not yet been demonstrated and there is an indication that a boiling-D_2O, metallic fuel reactor may have control problems under transient conditions. Metallic fuel elements may show economic promise with more advanced coolants such as H_2O, "fog," steam, gas, or organics.

The fuel element development program will accelerate materially when the HWCTR is placed in operation during the third quarter of 1961.

2. Reactivity Predictions

Although the criticality of full scale D_2O-moderated reactors with natural uranium can be assured, existing experimental data are inadequate for specification of optimum lattice configurations and for accurate predictions of attainable fuel exposure in some of the reactors of interest. Critical experiments with uranium metal and uranium oxide should provide needed data on cold, clean reactivity by the end of FY-1961. Data on the effect of fuel burnup on reactivity will be obtained from reactors now being built (viz., NPD-2 and possibly HWCTR). The latter information will supplement existing Canadian data on long-term reactivity changes, as derived from measurements on irradiated fuel samples.

3. Stability and Safety of Boiling Reactors

Preliminary calculations of transients in boiling reactors indicate that control of an oxide-fueled reactor will not be difficult. The control problem for a metal-fueled reactor has not been resolved yet and is a source of some concern. Void and temperature coefficients will be measured for verification of the calculations.

4. Unrecoverable Losses of D_2O

The results thus far of leakage measurements on samples of reactor components of conventional design show that it is possible to hold the unrecoverable losses of D_2O from these sources to a tolerable level. These results must be supplemented by experience with an operating reactor before a firm assessment of overall D_2O losses can be made. Data of the latter type will be obtained from the several experimental reactors now under construction. If necessary, the indirect cycle can be used to reduce D_2O losses, with only a small penalty in power cost.

5. Pressure Tubes

The limited information presently available is favorable in regard to the effect of prolonged irradiation on the mechanical properties of Zircaloy pressure tubes. One pressure tube at Hanford appears not to have deteriorated during about two years of irradiation. More data on this tube and on a tube that has been irradiated at Chalk River for three years should be available late in 1960.

6. Pressure Tube Joints

Various designs of mechanical joints for connecting zirconium pressure tubes to coolant distributors are being evaluated in other programs. The first performance data on the joints will be obtained from the PRTR and NPD-2 reactors. Excellent results have been obtained in out-of-pile tests of a joint which is made by bonding Zircaloy to stainless steel.

7. Heat Transfer and Hydraulics

Available experimental data on heat transfer of fuel assemblies that are cooled by boiling water are adequate to design a prototype reactor. More precise burnout and pressure drop data are required to attain the full potential of a power station reactor. An experimental program to better establish burnout limits has been initiated at Columbia University. The earliest data on full scale fuel element capabilities will come from operation of the HWCTR.

3. CURRENT STATUS OF CIVILIAN REACTOR PROJECTS

The heavy water-moderated power reactor program in the United States is closely coupled with the Atomic Energy of Canada Limited (AECL) program through an international agreement. In addition, considerable interest in heavy water reactors has been shown in several foreign countries. Table 3.1 illustrates this interest by summarizing pertinent data for known construction and study projects exclusive of those discussed in this report. Some of the data presented in Table 3.1 may be out of date, especially in the cases of foreign reactors.

Both the U.S. and Canadian programs are concerned primarily with natural uranium-fueled plants. However, prototype plants in the U.S. are using enriched fuel in order to reduce the reactor size. Both of the Canadian power reactors will be fueled with natural uranium. Current design studies and each of the plants which are either under construction or scheduled for construction are described briefly below and in more detail in the Appendix, Section 7. Their contribution to current technology is discussed in Section 5 and the plant economics, where applicable, is discussed in Section 4.

3.1 PLUTONIUM RECYCLE TEST REACTOR (PRTR)

The PRTR is a calandria type, through-tube reactor using Zircaloy-2 pressure tubes, natural UO_2 fuel with Pu-Al spikes, and pressurized D_2O coolant. It is being built to demonstrate the application of the plutonium recycle concept to various power reactor types; consequently, the coolant conditions are similar to those encountered in current power reactor technology. Due to the type of operation anticipated, no turbine-generator was installed and the reactor thermal power is dumped via a conventional heat exchanger system.

The PRTR development program has contributed substantially to current technology, especially in demonstrating the pressure tube concept of reactor core construction. In addition, development of swaged UO_2 fuel elements and D_2O sealing techniques are contributing to power cost reduction.

3.2 HEAVY WATER COMPONENTS TEST REACTOR (HWCTR)

The HWCTR is a pressure vessel, pressurized D_2O-cooled test reactor being built by du Pont at Savannah River. The core consists of 12 test fuel element positions surrounded by a ring of enriched fuel elements. The active core length (10 ft) is sufficient to test full scale fuel elements for natural uranium reactors.

3.3 CAROLINAS-VIRGINIA TUBE REACTOR (CVTR)

The CVTR is the only prototype or demonstration reactor construction project in the U.S. heavy water reactor program. The reactor core consists of internally insulated Zircaloy-4 U-tubes in a tank of cold D_2O moderator. The pressurized D_2O coolant is insulated from the moderator by Zircaloy shrouds which form several annuli of stagnant D_2O. This small (61.9 MW_{th}) reactor is enriched to ~2% U^{235}. The net plant rating of 17 MW_e is obtained with a fossil-fueled superheater.

Table 3.1 — Summary of Heavy Water Reactor Projects

Type of Reactor	BOILING D_2O									H_2O COOLED			PRESSURIZED D_2O			
Reactor	ANL 500 FC	ANL 500 NC	ANL 800 A	ANL 800 B	ANL 1000 MW Case 1	ANL 250 MW Case 2	ANL Case 3	Norway Halden	American Standard	NDA Liquid	NDA BSWR "Fog"	NDA SWR	American Standard	Sweden R3 ADAM Stage 1	Sweden R3 Stage 2	Sweden City of Vasteras
Coolant	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	90% D_2O 10% H_2O	D_2O	D_2O	H_2O	H_2O	H_2O	D_2O	D_2O	D_2O	D_2O
State of Coolant	Boiling	Boiling	Boiling	Boiling	Boiling	Boiling	Boiling	Boiling	Boiling	Liquid	Boiling	Steam	Liquid	Liquid	Liquid	Liquid
Pressure Container	Vessel	Vessel	Vessel	Vessel	Vessel	Vessel	Vessel	Vessel	Vessel	Tube	Tube	Tube	Vessel	Vessel	Vessel	Vessel
Moderator Temperature, °F	155	155	165	165	200	200	200	446	Avg. den. = 0.66 T avg. = 400		<212	<212	517	427.3	427.1	
Plant Size, net MW_e	280	310	325	370	248	62	62	None	250	100	200	200	250	9-11	31	None
Enrichment, % U^{235}	Natural	1 15	1.3	1.15/1.9	Natural	0.92	0.85	Spiked	1.3	Natural		Natural	1.1	Natural	Natural	Natural
Status	Conceptual design	Conceptual design	Conceptual design	Conceptual design	Conceptual design	Conceptual design	Conceptual design	Critical – June, 1959	Conceptual design	Conceptual design	Conceptual design	Conceptual design	Conceptual design	Under construction Startup 1960	Conceptual design	Under construction Startup 1960
Plant Cost Factors																
Steam pressure, psia	750	750	725	725	600	600	600	249	400	475	2000	2015	400	217.6	217.6	28.8
Steam temperature, °F	510	510	850	850	486	486	486	401	445	462	1050	1050	445	388.4	388.4	248
Boiler heat exchanger surface, ft^2/MW_e								$143/MW_{th}$						239.2		100
Reactor thermal power	986	1100	1000	1150	1000	250	250	10 (second core)	860	400	550	530	860	65	125	
Net plant efficiency, %	28.4	28 2	32 5	32 2	24 8	24.8	24.8		29	25.0	36 5	37.7	29	17	24.8	
No. of major cooling systems, including p p	1	1	1	1	1	1	1	2	1	2	1	1	2	2	2	2
Reactor outlet temperature, °F	510	510	850	850	486			446	445	500	1050	1050	535	420	428	284
Core tank volume, ft^3/MW_e	28.7	25.9	24 7	21.7	33.5			$\sim 98/MW_{th}$	32			43.8	26			
D_2O Cost Factors																
D_2O total inventory, tonnes/MW_e	0.405	0.366	0.349	0.306	0.44			Total = 16 T 1 6 tonnes/MW_{th}	0.85			1.3	0.73	3 64 (core and reflector)	2.08	0.26 t/MW_{th}
Bulk D_2O pressure, psia					600	600	600		400	Atm	Atm	Atm	1800	515	515	
Fuel Cost Factors																
Material	UO_2	UO_2	Boil zone – UO_2 Super zone – UO_2 + MgO	Boil zone – UO_2 Super zone – UO_2 + MgO	U metal	U metal	U metal	U metal	U-Zr-Nb alloy	U metal	UO_2	UO_2	U-Zr-Nb alloy	UO_2	UO_2	UO_2
Cladding	0.016 in. Zr-2	0 016 in Zr-2	Boil zone – 0.016 in. Zr. Super zone – 0.010 in. SS	Boil zone – 0.016 in. Zr Super zone – 0.010 in SS	0.020 in. Zr-2	0.020 in. Zr-2	0.020 in. Zr-2	Al	Zr-2	0.010 in. Zr-2	Center reg Zr-2 Outer reg. 316 SS	316 SS	Zr-2	0.031 in. Zr-2	Zr-2	0.0394 in. Al
Geometry	0 31 in diam. pins 85 pins/assembly	0.31 in diam pins 69 pins/assembly	Boil zone – 0.31 in. diam pins, 85 pins/assembly Super zone – 0 322 in. diam. pins, 85 pins/assembly	Boil zone – 0.31 in diam pins, 69 pins/assembly Super zone – 0.322 in. diam. pins, 85 pins/assembly	0.11 in. thick plates 9 plates/assembly. 12 ft long	0.11 in thick plates	0.11 in. thick plates	SS clad enriched UO_2 spikes	0.12 in. plates	Cored rods 0.500 in. OD 0 125 in ID 37-rod clusters	1½ in diam. rods	3/4 in diam rods	Plates 0 12 in. thick	0 669 in. diam. pellets 19-rod elements	0.669 in. diam. pellets 19-rod elements	0.622 in. diam. bars 7-bar elements
Tonnes nat. U/MW_e	0.1233	0.1304	0.1359	0 1155	0 159	0.158	0 158		0.21	0.459		0.45	0.214	1.84	0.535	0.0758/MW_{th}
Kg-U^{235} over nat./MW_e	None	0.556	0.922	0.692	None	0 323	0 22		1.3	None		None	0.85	None	None	None
Max. fuel temperature			1650							800		4770		2550	440.6	
Max. clad temperature			1200	1200						619		1155		446		305
Max. heat flux, Btu/ft^2-hr	400,000	373,000	242,000	390,000	400,000	400,000	400,000		-360,000	466,000				105,000	199,760	260,000
D_2O vol./fuel vol. in core										28.15				15.7	15.7	22.1
Initial Conversion Ratio	0.887	0 904	Boil 0.528 Super ht. 0.911	Boil 0.404 Super ht 0 723	0 901	0.869	0.766					0.72		0.77		
Cycling											Counterflow	Counterflow				
Fuel burnup, MWD_{th}/tonne	5000	6000	6000	6500	10,000	10,000	10,000		9000		10,000	5500		2800		
Fuel element heat transfer surface, ft^2/MW_e										92 9	83	105		~466	160	
Limitations Inherent in Reactor Type	← Plant capacity limited by size of available vessel → Thermal insulation between coolant and moderator (cold moderator cases)									Higher neutron absorption than D_2O Positive void coefficient			← Plant capacity limited by size of available vessels → Thermal insulation between coolant and moderator. Temperature rise limit. Large D_2O holdup in external system			

Table 3.1 — (Continued)

Type of Reactor	GAS COOLED					ORGANIC COOLED				LIQUID METAL COOLED		HOMOGENEOUS					
Reactor	Sweden ASEA	Russia Energiya Atomnaye	Czech	France EL-4	Du Pont – E-1	Canadian GE (OCDR)	Brown, Boveri	NDA (Proj. 3109) Case 1	NDA-SL	NDA SDR 10 MW_e	Czech	HRE-2	ORNL 2-Region Homogeneous	ORNL 1-Region Homogeneous	ASEA	Russia	B and W – NPG 2-Region Homogeneous
Coolant	CO_2	CO_2	CO_2	CO_2	He	Organic	Diphenyl	Santowax R	Organic	Sodium	Na	UO_2SO_4 in D_2O	UO_2SO_4-D_2O	UO_2-PuO_2 or UO_2-ThO_2	D_2O	D_2O	UO_2SO_4-D_2O
State of Coolant	Gas	Gas	Gas	Gas	Gas	Liquid	Liquid	Liquid	Liquid	Liquid	Liquid	Liquid	Liquid	Slurry	Homogeneous	Homogeneous boiling	Liquid
Pressure Container	Tube (bayonet)	Vessel	Vessel	Tube	Tube	Tube	Tube	Tube	Tube	Tube		Vessel	Vessel	Vessel			Vessel
Moderator Temperature, °F		158	158	—	120	120	176	150	155	130	158	~572	451	451		~400	
Plant Size, net MW_e	135	100-200	150 (gross)	80	100	150	32 gross	205	207	10		Produced 0.3 MW_e	105	300			150
Enrichment, % U^{235}	Natural	Natural	Natural	Natural	Natural		Natural	Natural	Natural	~3%	Natural		Full				100% U^{233}
Status	Conceptual design	Unknown	Under construction	Under construction	Conceptual design	Conceptual design	Conceptual design	Conceptual design	Conceptual design	Research and development	Conceptual design	Operating	Conceptual design	Conceptual design	Conceptual design	Conceptual design	Conceptual design
Plant Cost Factors																	
Steam pressure, psia		435	412		485	1600	580	900	185	850			215	215			620
Steam temperature, °F		750	750		670	780	590	750	375	850			338	338			
Boiler heat exchanger surface, ft^2/MW_e					283 + fins												
Reactor thermal power (MW)	444		590	260	377	465	100	700	990	46		5.22	450	1350			520
Net plant efficiency, %	30.4		25.4	30.7	26.5	32	32.0	29.3	20.9				23.3	22.2		24.3	28.8
No. of major cooling systems, including p.p.	2		2		2	2	2	2	2	3	3		2	2		2	2
Reactor outlet temperature, °F		790	800	970	715	800	615	800	500	950		572	482	482			572
Core tank volume, ft^3/MW_e			88		54.6	17		28	78.7				2.55				
D_2O Cost Factors																	
D_2O total inventory, tonnes/MW_e	0.556		0.32 (moderator only)	1.0	0.79	0.57		0.90	2.37		0.5-0.6		0.419	0.51		0.475	
Bulk D_2O pressure, psia	Atm	~900	880		400	Atm	Atm	Atm	Atm	Atm		2000	1000	1000		588	1500
Fuel Cost Factors																	
Material	UO_2	U metal	U metal	UO_2	U metal	UO_2	U metal	UC	U metal	UO_2		UO_2SO_4	UO_2SO_4	UO_2-PuO_2 or UO_2-ThO_2			$U^{233}O_2SO_4$
Cladding	0.039 in. S.A.P.	Be and Mg 0.018 in.	Mg-Be alloy	Be	Mg	Al	Mg	0.020 in. SAP + fins	0.025 in. Al	0.015 in. SS							
Geometry	0.160 in. diam. rods	0.16 in. [illegible] wire	Wire 4 mm diam.	Rods 0.51 in. diam.	0.09-in. thick ribbons	Concentric tubes	Tube	19-rod clusters of 0.540 in. clad OD	37-rod clusters of 0.50 in. OD rods	19-rod clusters		Solution in D_2O $9.6 \frac{gm\ U^{235}}{kg\ D_2O}$	Solution in D_2O $1.30 \frac{gm\ U^{235}}{liter}$ $\sim \frac{5\ gm\ total\ U}{liter}$	Slurry			Solution in D_2O core $0.95 \frac{gm\ U^{233}}{liter}$ blanket – ThO_2 pellets
Tonnes nat. U/MW_e	0.274		0.17	0.325	0.439	0.27	0.425	0.22	1.29	0.172	0.0286/MW_{th}					0.0946	
Kg-U^{235} over nat./MW_e	None	None	None	0	None		None	None	None	4.0	None						
Max. fuel temperature, °F					855	830		1442		4500	1202	572	482	482			
Max. clad temperature, °F	977	930	932	1110	842			850	570		1022						
Max. heat flux, Btu/ft^2-hr	152,200		74,200		78,000			225,000	177,000								
D_2O vol./fuel vol. in core	17.0							31.8	24.4								1.00
Initial Conversion Ratio					None		0.72	0.766		0.22							
Cycling					3000			4-zone radial shift, axial inversion	1/3 of fuel replaced each refueling								
Fuel burnup, MWD_{th}/tonne			3000	7-8000		5400		3900	~3000	5000 max						7900	
Fuel element heat transfer surface, ft^2/MW_e	12.1		118	130	326			129	257	99.3							
Limitations Inherent in Reactor Type	← [illegible] capacity [illegible] [illegible] size → ← Development of high-temperature, low absorption cladding and fuel [illegible] of pressure coolant safety. →				Compatibility of H_2 and Mg	Temperature limited by organic decomposition Temperature limited by coolant tube structural properties Power output limited by allowable amount of hydrogen in the core				Present technology of low absorption materials for high-temperature sodium containment in natural uranium cases		← Vessel corrosion →					

The CVTR is a prototype for a larger, natural uranium-fueled power plant. The current development program stresses the Zircaloy U-tubes and Zircaloy to stainless steel joints.

3.4 CANADIAN NUCLEAR POWER DEMONSTRATION REACTOR (NPD-2)

The NPD-2 is the only natural uranium-fueled power reactor that is nearing completion. It is a prototype (20 MW_e) of the Douglas Point Station, CANDU (200 MW_e). The reactor is a calandria-pressure tube type, in a horizontal position. On-power refueling is accomplished from both ends, providing virtually continuous countercurrent fueling. It is contended by AECL that this technique, which is complicated and requires reliable mechanisms, alleviates the control requirements substantially and reduces the maximum-to-average burnup ratio of spent fuel to almost 1.0.

The Canadian development program is directed exclusively toward natural uranium-fueled reactors. The main points of study are: Zircaloy pressure tubes, calandria fabrication techniques, D_2O seals and closures, pressure tube joints, refueling machines, fuel fabrication, and long term fuel burnup effects.

3.5 DOUGLAS POINT STATION (CANDU)

The Canadian full scale plant (200 MW_e) is presently scheduled for construction at Douglas Point, Ontario. It is similar to NPD-2 in orientation and mode of operation.

3.6 FWCNG GAS-COOLED REACTOR

The East Central Nuclear Group and the Florida West Coast Nuclear Group, under contract with the USAEC, are developing an advanced, high temperature, CO_2-cooled reactor plant. The current agreement provides for design of a 50 MW_e prototype of a 300 MW_e plant capable of operating on natural uranium fuel. Whether or not the prototype will be constructed has not been established. The General Nuclear Engineering Corporation is Nuclear Project Engineer and the American Electric Power Service Corporation is Principal Design Engineer. The prototype plant is described in Section 7.5.

Both the full scale and prototype plant concepts are characterized by high temperature coolant (delivered at 1050°F), high net efficiency (~34%), on-power refueling and low fuel cycle cost. In 1960 the prototype research and development program was interrupted and reoriented to replace the originally proposed stainless steel fuel cladding with beryllium, the same as would be required for the 300 MW_e plant. This work has already started with effort directed toward finned beryllium cladding, representing technology more advanced than any other known domestic or foreign beryllium development program.

3.7 DESIGN STUDIES

Several design studies of power reactors have been carried out which have not yet led to construction programs.

The du Pont Company conducted a design study in parallel with, but independent of, a joint Sargent & Lundy, Engineers – Nuclear Development Corporation of America (S&L-NDA) study. Both of these studies[7,43,46] indicated that the boiling D_2O-cooled, cold D_2O-moderated, pressure tube type reactor in a direct cycle plant had the most promise with 1959 technology. In the intervening year, all three companies have continued investigations of various phases of the technology. SL-1776, a comparison of the direct and indirect cycles for identical boiling D_2O-cooled reactors for a 200 MW_e plant, shows that there is no great economic difference between them, the difference in power cost being only 0.11 mill/kwh. The selection of the direct cycle case was influenced by the belief that D_2O leakage could be controlled, and that when adapted to fog or steam cooling with superheat the direct cycle would be more economical than the indirect cycle. The use of droplet (fog) cooling was also studied briefly and is shown as the potential coolant for the reactor. A more detailed investigation of fog cooling in an H_2O-moderated reactor is being carried out by NDA and CISE under USAEC-Euratom sponsorship.

4. ECONOMIC STATUS

The cost estimates presented in this section represent the current economic status of the heavy water-moderated power reactor program. Available cost data for the construction and study projects are summarized in the Appendix, Section 7. The pertinent technical data describing each construction project or design study concept is also discussed in Section 7 and summarized in Table 7.1.

To permit economic evaluation of the different D_2O-moderated power reactor design concepts, it has been necessary to normalize the reactor designs as well as the cost data to a common basis. The previous conclusions reached by du Pont, and by Sargent & Lundy and NDA independently have again been confirmed; namely, that of the various D_2O-moderated power reactors, the boiling D_2O-cooled plant yields the lowest power cost for units of large capacity (300 MW_e). Therefore, cost estimates have been prepared for this reactor concept, based on the latest AEC ground rules, for both 100 and 300 MW_e nominal size power plants. These estimates reflect the current economic status of this reactor concept. Potential cost reductions, based on success of the research and development program, are described in Section 6.

4.1 CURRENT COST ESTIMATES – BOILING D_2O, DIRECT CYCLE PLANTS

Cost estimates have been prepared for both 100 and 300 MW_e nominal size reactor plants of the latest boiling D_2O-cooled, direct cycle design. The estimate for the 300 MW_e plant is based on the design concept and cost data reported in Reference SL-1815. This study and SL-1661 indicate that an average fuel burnup of 8500 MW-d/metric ton is attainable with natural UO_2 fuel using a four-zone refueling scheme. To conform to an additional ground rule that the maximum burnup of natural UO_2 shall not exceed 8500 MW-d/metric ton, power costs for an average burnup of 7500 MW-d/metric ton are also shown. The costs for the 100 MW_e plant are based on a scaled version of a reactor design concept plant reported in Reference SL-1815 and a prototype plant reported in SL-1773. The power cost estimates are summarized in Table 4.1. The reactor description and plant characteristics are presented in Section 7.1. Capital costs, in accordance with the USAEC system of accounts, are given in Table 7.4 of the Appendix.

Pressure tube reactors have several unique features which affect power cost and should be explained. Each pressure tube is a closed unit, the top being a shielding plug which must be removed to gain access to the fuel channel contents. Fuel elements can be removed by lifting vertically by means of a refueling machine. The two section fuel element can be repositioned radially and axially without difficulty. The operation consists of removing the top segment and inserting it in the desired channel, followed by the identical operation on the lower segment. Rotating or inverting the fuel element is not required. In this way, the highly burned center section of the fuel channel is repositioned towards the two ends of the core and comparatively fresh fuel is located around the reactor midplane.

Since each pressure tube-fuel element assembly is essentially independent of all others, either on-power or off-power refueling is feasible. In the analysis of the boiling D_2O-cooled reactors

Table 4.1 — 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

presented herein, off-power refueling has been assumed. Studies in 1959 by NDA and S&L, reported in SL-1661, indicated that most of the fuel burnup advantages of on-power refueling can be obtained with a four to six-zone radial shift refueling scheme where the two-section fuel element is repositioned axially, if desired. Reactor downtime required for multizone refueling is not a major factor in pressure tube reactors since the difficult operation of removing the pressure vessel head and other equipment is not necessary. Also, shutdown and startup times are not controlled by thermal shock. The refueling scheme assumed in these studies, that is, four-zone radial shift, does not exceed the 10% downtime specified in the ground rules.

4.2 EFFECT OF ECONOMIC GROUND RULES

The latest AEC ground rules for evaluating central station power reactor plants have been applied. The cost of land and land rights and off-site improvement costs for the recommended Massachusetts site have been incorporated.

The economic basis upon which a nuclear power station is selected has a major influence on the type of plant which will produce minimum power cost.

To illustrate the effect of some of the ground rules on power cost, several of the more important factors are discussed below. It should be noted that no change in concept has been considered and that the effects shown might be increased by a re-evaluation of the reactor.

4.2.1 Heavy Water Costs

The cost of the inventory of heavy water, currently evaluated at $28/lb, is treated as a capital charge presently evaluated at 12.5%/yr. If it were possible to lease D_2O from the AEC at 4.0%/yr, as is the case with uranium fuels, the power costs for the 100 and 300 MW_e plants would be reduced by approximately 1.43 and 0.87 mill/kwh, respectively.

4.2.2 Fuel Reprocessing Costs

The value of spent fuel from enriched reactors is generally so great that it is an economic necessity to reprocess the fuel for recovery of uranium and plutonium. Therefore, the cost of reprocessing is an important factor in the overall economics of these reactors. Reprocessing costs are not so important for D_2O-moderated reactors because the initial cost of natural uranium is relatively low.

Published AEC data on fuel costs for power reactors are predicated on reprocessing of fuel.[17] It can be shown from these data that the fuel cost for enriched reactors would generally increase 1.0 to 2.5 mills/kwh if the spent fuel were not reprocessed. If reprocessing were not considered for the 100 and 300 MW_e boiling-D_2O, direct cycle plants, increases of only 0.20 and 0.41 mill/kwh, respectively, would be expected in the total fuel costs. Therefore, the natural uranium-fueled, D_2O-moderated reactors are not strongly influenced by the availability of a fuel reprocessing industry.

4.2.3 Uranium Price

The current AEC price for natural uranium is $40.50/kg of U; this price has been used in fuel cost estimates in this report. However, the free market price of natural UO_2 is substantially lower. Recent Canadian data indicate that this cost is $30.10/kg of U. If this UO_2 cost were applied to 100 and 300 MW_e plants, fuel cost reductions of 0.25 and 0.20 mill/kwh, respectively, would result.

4.2.4 Capital Charges

The current AEC ground rules call for a 14% capital charge on plant investment. This rate is typical of privately-owned utility systems and places a premium on minimizing capital investment. Canadian estimates for D_2O-moderated reactors employ capital charges representative of those used by public utilities. These rates place a premium on minimizing fuel costs and allows the acceptance of higher capital investment to achieve this objective at no net power cost penalty. (See Appendix, Section 7.7 for amplification.)

4.3 SUMMARY OF PREVIOUS STUDIES

4.3.1 D_2O-Cooled, D_2O-Moderated Design Concepts

During the course of the heavy water power reactor program, du Pont, Sargent & Lundy, and NDA have prepared a number of cost estimates for D_2O-moderated-and-cooled power reactor plants of capacities from 70 to 460 MW_e. To permit an economic comparison of the different concepts, cost estimates for both natural uranium metal-fueled and oxide-fueled plants were prepared on a common basis. A nominal 300 MW_e reactor power plant was selected as the base size for cost data normalization. Estimates have been prepared for the following reactor concepts which could be constructed within the framework of current technology:

1. Boiling D_2O-cooled, direct cycle, pressure tube, cold moderator, natural uranium oxide-fueled. Reference 44.
2. Liquid D_2O-cooled, pressure vessel, hot moderator, natural uranium metal-fueled. Reference 46.
3. Liquid D_2O-cooled, pressure tube, cold moderator, natural uranium metal-fueled. Reference 46.
4. Liquid D_2O-cooled, pressure tube, cold moderator, natural uranium oxide-fueled. Reference 46.

The design details for these reactor concepts are discussed in SL-1773 and the estimated power costs are summarized in Table 4.2. The summarized power costs include effects resulting from studies presently being conducted, and are therefore complementary to the costs reported in References DP-480 and SL-1773. Ground rules used as the basis for establishing these costs were:

Plant load factor – 80%
Fixed charges for depreciation – 14%/yr
Fixed charges for D_2O inventory – 12.5%/yr
Fixed charges for fuel fabrication – 12%/yr of inventory value, excluding fuel value
Uranium use charge – 4%/yr of fuel material inventory value
D_2O cost – \$28/lb
Fuel fabrication costs (exclusive of uranium cost) – \$25/kg U for uranium metal
\$64/kg U for uranium oxide

These rules, as well as the computational methods used, differ slightly from the latest AEC ground rules, as tabulated below. Therefore, the power cost estimates for any single concept are not intended to be indicative of individual performance. Since the differences in ground rules would affect all concepts in the same way, the increments between power costs for different concepts present a fair evaluation of comparative economic performance.

Table 4.2 — Summary of Capital and Operating Costs – 300 MW_e, D_2O Reactor Evaluation

(For comparison purposes only)

Reactor Power Plant Concept	Plant Capital Cost, mills/kwh	Heavy Water Inventory, mills/kwh	Total Fuel Cost, mills/kwh	D_2O Makeup, Operation, Maintenance, and Insurance, mills/kwh	Total Power Cost, mills/kwh	Total Power Cost Differential, mills/kwh
Boiling D_2O-cooled, pressure tube, direct cycle, cold moderator, natural uranium oxide-fueled[44]	4.6	1.3	1.9	1.0	8.8	—
Liquid D_2O-cooled, pressure vessel, indirect cycle, hot moderator, natural uranium metal-fueled[46]	5.1	1.1	2.4	1.1	9.7	0.9
Liquid D_2O-cooled, pressure tube, indirect cycle, cold moderator, natural uranium oxide-fueled[46]	5.7	1.5	2.3	1.1	10.6	1.8
Liquid D_2O-cooled, pressure tube, indirect cycle, cold moderator, natural uranium metal-fueled[46]	5.4	1.2	2.1	1.1	9.8	1.0

Cost data for metal-fueled plants are on the basis of an average fuel exposure of 3300 to 5100 MW-d/metric ton-U, which corresponds with 100% batch discharge. Cost data for oxide-fueled plants are on the basis of an average fuel exposure of 7800 to 8800 MW-d/metric ton-U, which requires a multizone fuel discharge.

Ground Rule Variations

Cost Item	Comparative Cost Estimate	AEC
Capital		
Site	Northeast U.S. site	Western Massachusetts site
Step-up transformer	Included	Not included
General and administrative expense	Part of direct cost	Indirect cost as percentage of direct cost
Indirect capital costs	4% escalation per year included	Escalation not included
	15% top charges including engineering, design, and inspection plus interest during construction	14.6% engineering, design, and inspection 8.1% interest during construction
Fuel costs	Includes 12%/yr inventory value excluding fuel value	Non-nuclear fuel inventory interest charge included as part of working capital, not as a fuel cost
	Fuel value inventory charge based on initial fuel value	Fuel value inventory charge based on weighted value as fuel is consumed
Working capital	Not included as separate item	Included
Insurance	Includes third party liability, government indemnification, and all risk nuclear property insurance	Includes only third party liability and government indemnification. All risk insurance included as part of 14% capital charge

Estimates for the liquid D_2O-cooled reactors as previously reported in DP-480 remain unchanged. However, the boiling D_2O concept design has been updated recently in studies performed by NDA and S&L (Reference SL-1815). As a result, two major changes have been made:

1. The cold feedwater return has been routed directly to the steam drum, thereby increasing the available recirculating pump net positive suction head and permitting a more desirable arrangement of the reactor plant equipment.
2. The steam plant regenerative heating feedwater cycle has been revised to incorporate subcoolers in each of the feedwater heaters in conjunction with pumped instead of cascaded drains. In this manner, net plant efficiency has been increased from 27.9 to 28.6%.

4.3.2 Direct and Indirect Cycles with a Boiling D_2O Reactor

Sargent & Lundy and NDA recently completed a technical and economic evaluation of direct and indirect steam plant cycles in conjunction with the boiling D_2O-cooled, pressure tube, cold moderator, natural uranium oxide-fueled design concept.[43] The reactor arrangement and design parameters correspond to those represented for the 200 MW_e direct cycle plant in References SL-1565 and SL-1565, Addendum No. 1. However, in the case of the indirect cycle, the plant arrangement was modified to incorporate the H_2O steam generator and the steam plant was modified for a more conventional H_2O turbine and regenerative feedwater heater cycle. This study indicates that the two cycles are comparable from a technical standpoint, with the direct cycle indicating a slight economic advantage. Plant characteristics are summarized in Table 7.1 of the Appendix and the cost comparison, based on the previously discussed ground rules, is summarized in Table 4.3. Complete details of the comparative study are reported in Reference SL-1776, NDA 2131-6.

The economic advantage of the direct cycle plant in comparison to the indirect cycle is due to the benefit of higher temperature and pressure steam, which compensates for the smaller D_2O inventory and greater efficiency of the H_2O steam plant. While this economic difference for the current technology 200 MW_e plant is very slight (0.11 mill/kwh), the differential should be increased in future plant designs of more advanced concepts such as fog cooling. However, should the unrecoverable D_2O losses be less than the 2% per year allowance made in the economic analyses, the direct cycle will improve by comparison.

Table 4.3 — Comparative Summary of Capital and Operating Costs for 200 MW_e, Boiling D_2O-Cooled, Pressure Tube, Natural Uranium Oxide-Fueled Power Reactor – Direct vs Indirect Cycle

	Indirect Cycle 1567×10^6 kwh/yr at 0.8 Operating Factor			Direct Cycle 1570×10^6 Btu/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						
Plant investment	62.050	8.686	5.547	57.141	8.000	5.097
D_2O inventory investment	12.970	1.620	1.034	16.500	2.060	1.310
Total investment	75.020	10.306	6.581	73.641	10.060	6.407
Operating						
Fuel cost		3.336	2.133		3.336	2.124
D_2O makeup		0.213	0.136		0.336	0.214
Operation, maintenance, and insurance		1.646	1.050		1.646	1.047
Total operating costs		5.195	3.319		5.318	3.385
Total capital and operating costs		15.502	9.900		15.378	9.792

5. CURRENT TECHNOLOGY AND RESEARCH AND DEVELOPMENT PROGRAM

This section discusses major problem areas associated with natural uranium-fueled, D_2O-cooled-and-moderated power reactors, in particular, the boiling D_2O-cooled, pressure tube type. The technology of the FWCNG gas-cooled reactor is discussed in another volume of this program report under the general heading of gas-cooled reactors. D_2O-moderated reactors cooled by other fluids such as organic or H_2O fog are not currently under active development, but will be the subject of comparative evaluation studies during the course of this program to assess any new developments in the technology of such systems.

5.1 PHYSICS

5.1.1 Introduction

The major physics problems for heavy water-moderated power reactors fueled with natural uranium involve improvement in the accuracy of predicting the cold, clean reactivity, reactivity coefficients, and the long-term reactivity changes for lattice configurations of interest. In the design of a natural uranium-fueled reactor, the expedient of increasing enrichment to overcome uncertainties in reactivity predictions is not available and there is a greater need for accuracy in the prediction methods. There is no doubt, from the reactivity standpoint, that a D_2O-moderated, natural uranium-fueled power reactor can be built and operated. The question is rather one of specifying the optimum lattice and refueling scheme which will provide high fuel exposures and, consequently, minimize power costs. Critical experiments in progress should provide most of the needed data on cold, clean reactivity and on reactivity coefficients by the end of FY-1961. New data on the effect of fuel burnup on reactivity will be obtained from reactors now being built (viz., NPD-2 and possibly HWCTR).

5.1.2 Current Status of Technology

Measurements of Cold, Clean Reactivity

Measurements of buckling have been obtained in critical, exponential, and substitution experiments. Results of tests over a fairly wide range of lattices are summarized (except where noted) in Reference 6. Table 5.1 gives the range of these measurements with respect to values of fuel area per element and equivalent cell radius (calculated on an equal area basis).

A limited set of measurements of k_∞ have been made in the PCTR at Hanford on oxide lattices in D_2O.[21] A 19-rod lattice (3.78 in.^{2} per element) was measured at lattice pitches of 7, 8, and 9 in. (equivalent cell radii of 3.7, 4.2, and 4.7 in.) Tests in the PLATR at NDA have begun recently to determine k_∞ for a number of test lattices of interest to the present program.

Table 5.1 — Range of Buckling Measurements

Type of Rod	Fuel Area per Element, in.^{2} Min.	Fuel Area per Element, in.^{2} Max.	Equivalent Cell Radius, in. Min.	Equivalent Cell Radius, in. Max.
Single metal rod	0.5	3.1	2.0	6.8
Single oxide rod	1.1	2.6	3.3	5.6
Clustered metal	1.4	4.0	2.5	6.0
Clustered oxide	2.0	6.6	2.7	6.8

As part of the k_∞ measurements and of some of the critical and substitution experiments, thermal flux traverses have been made to aid in the calculation of the thermal utilization. A few measurements of the ratio of the fast radiative captures in U^{238} to the thermal captures have also been made. These aid in the calculation of resonance escape probability. From measurements[22] of the ratio of U^{238} fissions to thermal fission in U^{235}, values of the fast fission factor may be calculated. The age in D_2O has been measured both for pure D_2O and as a function of the amount of H_2O impurity.[23]

Measurements of Reactivity Coefficients

The reactivity effect of removing the coolant from a lattice has been measured in the PCTR at Hanford,[21] in substitution experiments in France (Aquilon facility),[24] and by measurements of buckling in the PDP and SE.[25] A number of measurements of moderator temperature coefficient have also been made in the PDP and SE.

Recent work on the fuel temperature coefficient of reactivity (Doppler broadening of the U^{238} resonances) has included: (1) a corrected set of measurements of the temperature coefficient of the resonance integral obtained from activation measurements with single rods of U metal and UO_2, and (2) a calculated value of the temperature coefficient of the resonance integral obtained from the change in k_∞ with fuel temperature for a 7-rod UO_2 fuel element in the PCTR. The above measurements were obtained with a uniform fuel temperature.

Reactivity vs Burnup

To date, there have been no reported U.S. measurements of long-term isotopic concentration or reactivity changes for single or clustered UO_2 rods in D_2O moderator. However, data have been reported for Windscale slugs and NRX slugs of natural uranium metal. The Windscale slugs were irradiated[26] to an exposure of approximately 1000 MW-d/metric ton-U, while the NRX slugs were irradiated[27,28] up to ~3000 MW-d/metric ton-U. These irradiations were performed in lattices of single rods of uranium metal arranged in their respective moderators of graphite and D_2O.

Isotopic concentrations and effective cross sections have been determined for the NRX slugs. Plutonium and uranium concentrations were measured by means of gravimetric, volumetric, spectrophotometric, and alpha-counting techniques.[29]

Measurements of long-term reactivity have been made at Harwell,[26,30,31] Argonne National Laboratory,[32] and at Chalk River.[33] Most of the data are for the slugs irradiated at NRX, although

Littler's measurements[31] were on slugs that had been irradiated at Windscale to exposures up to about 200 MW-d/metric ton-U. The Chalk River reactivity measurements were made by varying the D_2O moderator and reflector height, while oscillator and danger coefficient measurements were made at Harwell and Argonne, respectively.

Measurements of the change in the thermal regeneration factor, η, in the terms of the change in equivalent U^{235} concentration and boron poisoning (representing fission product poisoning) have been made recently in the Reactivity Measurement Facility at the MTR. The data were obtained for rods of natural UO_2.

Calculation Techniques

Cold, Clean Reactivity and Reactivity Coefficients

There are several semi-empirical methods of calculating the buckling of D_2O-moderated lattices. These methods are described in Reference 6. Table 5.2, taken from this reference, gives a brief description of the methods. All of the methods have been fitted in some degree to the experiments which they were designed to calculate. The range of confidence of three of the most successful calculation techniques is presented in Table 5.3. Within this range, these methods predict the reactivity of the more precise experiments to better than 1%. This means that measured bucklings are predicted to within about ±25 μb. However, if these methods are compared outside their range of validity, the agreement between them is poor. For example, for a 37-rod cluster of 0.5 in. UO_2 rods, the French and Swedish methods lead to reactivity predictions (k_{eff}) that differ by as much as 2% for a 16 in. pitch. This difference implies that the internal details of the calculation methods do not describe correctly the physical processes involved. Extrapolations of present data on the basis of these methods are thus open to question.

Table 5.3 — 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[7,8]	8.8	—	11.1	—
Du Pont 1K-300[9]	—	3.1	—	8.5

Semi-empirical calculations have been matched to experimental reactivity coefficients with some success, e.g., change of reactivity with loss of coolant. However, this may again be attributed to the fact that the available experimental data were used in the development of the methods, and extrapolations may be risky.

The calculation of the fuel temperature coefficient of reactivity (Doppler broadening of U^{238} resonances) has been based on experimental values of the temperature coefficient of the resonance integral for the case of uniform fuel temperature. The effect of nonuniform fuel temperature, which is difficult to study in a critical facility, has been treated by running a number of Monte Carlo calculations for a single UO_2 rod at various temperatures. The results were used to obtain an approximate formula for the temperature coefficient of the resonance integral for a rod temperature distribution corresponding to uniform heat generation and thermal conductivity.

Table 5.2 — Semi-Empirical Methods for Calculating D_2O Lattices[6]

Parameter \ Approach	Canadian	French	Savannah River	Swedish	Other	Notes
Fast effect (ϵ)	Spinrad (modified) 1. Considers 3 neutron energy groups. 2. For clusters (internal moderation), Spinrad cross sections modified to yield experimental fast to thermal fission ratios.	Spinrad (modified) 1. Considers 2 neutron energy groups. 2. Treats internal moderation with no adjustment to Spinrad's cross sections.	Spinrad (modified) 1. Considers 3 neutron energy groups. 2. Spinrad's cross sections used. ϵ does not treat internal moderation. See p, below. 3. ϵ does not include fast radiative captures in fuel.	Carlvik and Pershagen 1. Considers 3 neutron energy groups. 3rd group quite different from 3rd group in Spinrad. 2. Uses measured cross sections (BNL-325) and treats internal moderation.	Spinrad 1. Considers 3 neutron energy groups. 2. Does not treat internal moderation; e.g., neutron must escape to surrounding moderator to be considered in ϵ.	
Thermal utilization (f)	Kushneriuk 1. $\frac{\varphi_s}{\varphi_f}$ based upon integral transport theory result for rod blackness. 2. $\frac{\varphi_m}{\varphi_f}$ uses diffusion theory with transport theory boundary condition for λ.	Amouyal, Benoist and Horowitz 1. $\frac{\varphi_s}{\varphi_f}$ based upon integral transport theory (modified P_3) 2. $\frac{\varphi_m}{\varphi_f}$ similar to Kushneriuk with a refinement on λ computation.	P_3 Approximation 1. Use of P_3-approximation to one velocity transport formulation.	Pershagen and Carlvik 1. $\frac{\varphi_s}{\phi_f}$ based upon an S_4 approximation to one velocity transport theory. 2. $\frac{\varphi_m}{\varphi_f}$ as per Kushneriuk.	Bessel Function Lattice Sum Technique (Hanford) 1. An exact calculation involving the sum of a lattice array of zero order Bessel functions of the second kind.	$\frac{\varphi_s}{\varphi_f} = \frac{\text{Surface flux}}{\text{Average flux in fuel}}$ $\frac{\varphi_m}{\varphi_f} = \frac{\text{Average moderator flux}}{\text{Average fuel flux}}$ λ = Extrapolation length in strong absorber
Resonance escape probability (p)	Critoph 1. Assumes resonance absorption takes place at single energy; expresses p in linear form. 2. Resonance integrals use Hellstrand's measured results with $(S)_{eff}$ modified from Swedish. 3. Employs ω-correction based upon extrapolated experimental results for fast flux distribution.	Empirical 1. Uses elementary exponential formulation. 2. Resonance integrals use adjusted constants indirectly determined from French buckling measurements. $(S)_{eff}$ similar to Critoph. 3. Employs ω-correction with refined theoretical fast flux distribution.	Empirical 1. p expressed in linear form as in Canadian method. 2. Resonance integrals normalized to SRL experiments. Empirical integral formulations thus account for inadequacies of models for other parameters (e.g., ϵ). $(S)_{eff}$ as per Critoph. 3. p includes fast radiative captures in fuel.	Hellstrand 1. Uses elementary exponential formulation. 2. Resonance integrals per Hellstrand based on measured values of integrals. $(S)_{eff}$ per Pershagen and Carlvik. 3. Employs ω-correction as per Critoph.	"Monte Carlo" 1. Method using historical probability approach.	$(S)_{eff}$ = Effective surface for cluster. ω = Nonuniform slowing down in moderator
η	1. Cross section value.	1. Value determined from French experiment.	1. Cross section value.	1. Cross section value.		
Thermal diffusion area (L^2)	1-2. Transport mean free path and absorption cross section consider all components in cell. 3. f in computation as above.	1. Transport mean free path for cell considers only fuel and moderator. 2. Absorption cross section for cell weighted for all components. 3. f in computation as above.	1-2-3. Flux-weighted average transport and absorption cross section evaluated as in P_3 calculation for f. Averages include all materials in lattice cell.	1-2. Same as Canadian method. 3. f in computation as above.		All subject to Behrens corrections.
Fast diffusion area – age (τ)	1. Slowing down in all cell materials considered. 2. Two types of neutrons considered: those which emerge from fuel without an inelastic collision, and those which leave after an inelastic collision. 3. L^2_{sm} = 120 cm^2.	1. Slowing down in other than moderating material or fuel neglected. 2. Two types of neutrons considered: those which emerge from fuel without an inelastic collision, and those which leave after an inelastic collision. 3. L^2_{sm} = 120 cm^2.	1. Slowing down in other than moderating material or uranium neglected. 2. Uses only one type of neutron. Inelastic scattering in uranium taken into account by assuming uranium to be 1/2 as good a moderator as D_2O. 3. L^2_{sm} = 120 cm^2.	1-2. Similar to French method. 3. L^2_{sm} = 120 cm^2, with correction for moderator temperature.		All subject to Behrens corrections. L^2_{sm} = slowing down area in pure D_2O for a non-inelastically scattered neutron (20°C).

Reactivity vs Burnup

The close agreement between experimental and theoretical effective cross sections and isotopic concentrations for Pu^{239}, Pu^{240}, and Pu^{241} has increased the confidence in long-term reactivity predictions for single-rod type, uranium metal-fueled D_2O-moderated lattices. Effective capture cross sections, calculated via the method of Kushneriuk,[34] agree well with data obtained from experimental analyses. Experimental values of plutonium isotopic concentrations as derived from mass spectrometer techniques are in very good agreement with calculated values.[35] However, there is need for improvement in the prediction of reactivity change with burnup. Extension of the calculation techniques to cluster rod elements with moderating coolant is necessary.

Refueling Schemes

The attainment of high average burnup of the fuel in a natural uranium reactor requires the use of refueling schedules in which the fuel is relocated a number of times during its stay in the reactor. With methods such as countercurrent refueling or radial-shift refueling (with optional axial inversion of the fuel during the shift), the calculated average burnups are as high as three times that obtained with simple batch refueling. The final choice must be made on the basis of a compromise between attainable burnup, control requirements, operational complexity, the resulting reactor power pattern, and total reactor downtime for refueling.

To calculate the fuel exposure that can be achieved by use of a given refueling plan, values of long-term isotopic concentration and reactivity changes for the lattice are used in conjunction with reactor equations describing the particular refueling scheme. Preliminary calculations have been made on the assumption that the power pattern is not affected by burnup. These calculations indicate that average fuel exposures of about 7500 MW-d/metric ton-U are possible in a 200 MW_e boiling reactor with only a few shifts of each fuel element during its lifetime. Limited additional calculations show that the power pattern may change unfavorably with burnup for some refueling schemes. Consequently, further calculations are in progress to include the effects of changes in the power pattern on the burnup and power limits of the reactor.

5.1.3 Research and Development Program

The principal objective of the experimental and theoretical program is to obtain greater accuracy in the predictions of (1) initial reactivity, (2) reactivity coefficients, and (3) long-term reactivity and power pattern changes for reactor designs and refueling schemes suitable for economical power production. As discussed in Section 5.1.2, insufficient experimental data are available at the present time to obtain reasonably accurate predictions. The proposed program will supply this missing information and permit a more confident choice of the final lattice configuration, optimized with respect to power costs, control, and safety.

The Process Development Pile (PDP) at the Savannah River Laboratory has been converted to a large critical facility to permit accurate buckling measurements for a variety of natural uranium, D_2O lattices. The current and future tests will include large variations in number and size of fuel rods and in lattice pitch. Initial tests have been made with 1 in. diameter, uranium metal rods. Additional tests are planned with UO_2 rods. Following these tests, both the PDP and PSE will be used to measure the buckling of engineering lattices (i.e., lattices with process and calandria tubes) with both uranium metal and UO_2 rod clusters.

In conjunction with the above experiments, a series of k_∞ measurements will be made in the Pawling Lattice Test Rig (PLATR) at NDA. The PLATR, which was designed to permit rapid measurements with a small sample of the lattice of interest, will be used to test a number of engineering lattices. The PLATR will be used to investigate the effect on reactivity of rod size and number, rod spacing, cladding material, coolant worth, and other engineering factors. Comparison of results from PLATR and PDP for the same lattices should also provide information on the migration area.

During the course of this experimental program, special measurements will be made to obtain thermal flux traverses, the ratio of fast radiative captures in U^{238} to thermal captures, and the ratio of U^{238} fissions to fissions in U^{235}. These measurements will aid the supporting theoretical work for predicting both initial reactivity and long-term reactivity changes. In addition, studies of the coolant void coefficient and the temperature coefficients will be made in the PSE and PLATR.

Data on the changes in reactivity and isotopic concentration with burnup will be obtained from irradiation of elements in reactors now under construction (e.g., NPD-2, PRTR, and HWCTR). The first meaningful data of this type for reasonably long irradiation times (e.g., greater than 6000 MW-d/metric ton U) will probably not be available until 1963. Clustered rod elements of UO_2 will be irradiated in the HWCTR shortly after startup in late 1961. It is planned to obtain some information on long-term reactivity changes while the elements are in the HWCTR. Measurements of the change in reactivity with burnup in the NPD-2 will provide an important source of data pertinent to the D_2O-moderated power reactors, particularly since NPD-2 is fueled completely with natural UO_2. Data of this type may serve as a means of normalizing calculations to improve accuracy in predictions. However, in view of the uncertainties associated with such in-pile tests, separate out-of-pile tests on reactivity will probably be necessary. Accurate reactivity measurements could be obtained from buckling measurements or from k_∞ measurements in a facility such as PLATR, if modified to handle irradiated fuel elements. Such tests, in conjunction with measurements of isotopic concentrations, would provide valuable data for improvement or verification of the prediction methods.

From the results of the above experimental work and the theoretical studies discussed below, the design of the prototype reactor will be selected and a full scale critical experiment conducted in the PDP. This experiment would include measurements of flux pattern as a function of control rod position, control rod worth, and cold, clean reactivity.

The first phase of the theoretical program will be concerned with the development of methods for accurately predicting cold, clean reactivity and reactivity coefficients. Studies directed towards the development of both theoretical and semi-empirical methods for predicting these properties have been conducted by NDA and the du Pont Company. This work will be enhanced appreciably as the result of the new physics data to be obtained under the experimental program.

Work during the summer of 1960 will be centered on the selection and initial firming up of semi-empirical methods for calculating properties for the power reactor types of lattices. As more experimental data become available, the methods will be modified and improved. The methods should be sufficiently accurate for selection of the prototype reactor by the end of FY-1961.

As a backup program to the semi-empirical methods approach, work on more fundamental calculation methods should be continued to improve understanding of the physics of D_2O-moderated reactors with clustered rod elements and moderating coolant. The studies would be concerned with both initial reactivity and the effects of burnup. This basic approach may be particularly significant in serving as a guide to more accurate burnup calculations, since irradiation data for clustered rod fuel elements will be quite limited for several years.

Finally, improved methods for calculating the average fuel exposure and the variation in reactor power pattern and control rod position with burnup will be developed for the refueling schemes and reactor conditions of interest to D_2O-moderated power reactors.

5.2 FUEL AND MATERIALS

Two materials, uranium oxide and uranium metal are being developed as alternative fuels for a 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 at Shippingport 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. In addition, reactor stability considerations may preclude their use in boiling water reactors. An extensive program to resolve this question has been undertaken.

5.2.1 Uranium Oxide

Uranium oxide fuel, in the form of sintered pellets, has been proven to a relatively high degree as the result of extensive in-pile testing, primarily by Westinghouse, General Electric, and AECL.[1,18] It is significant that the PWR Core I blanket, consisting of sintered pellets of UO_2 sheathed in Zircaloy-2, has achieved greater than 10,000 MW-d/metric ton-U burnup in peak regions; there have been only one or two suspected failures among 95,000 rods. The irradiation experience to date indicates that sintered oxide pellet elements may withstand exposures of at least 25,000 to 30,000 MW-d/metric ton-U and should, in general, perform as follows:

Temperature Limit. A small degree of melting of the oxide probably is not detrimental to fuel element integrity. Some of the Westinghouse specimens did not fail when as much as 42 volume percent of the oxide was molten. Since other specimens with molten zones did fail, the melting temperature (~5000°F) is generally accepted today as a thermal design limitation. The maximum design temperature is usually limited to about 4000°F to provide a safety margin between normal operating temperatures and the melting point.

Power Limit. Experience with the release of fission gas as a function of power output indicates that sintered pellets of UO_2 can be designed for a nominal maximum rating of about 15 kw/ft of length; the NPD-2 is being designed for 13.5 kw/ft. An empirical correlation of fuel performance that was developed by the Canadians on the basis of experimental irradiations shows that grain growth occurs at a power level of about 12 kw/ft, and that central melting begins at about 20 kw/ft.

High-Burnup Swelling. The Westinghouse data indicate that no detectable swelling occurs in the PWR-type elements at burnups somewhat in excess of 10,000 MW-d/metric ton-U unless considerable center melting or water-logging occurs. A number of rods with molten zones increased in diameter only a few percent. These results indicate that sintered pellets of at least 93% of theoretical density retain over 99% of the gaseous fission products within the oxide when operating at less than 2700°F. The interstices and pores in the oxide can accommodate fission products corresponding to a quite high burnup (perhaps 30,000 to 50,000 MW-d/metric ton-U) without distorting the oxide body.

Consequence of Failure. Data from irradiation tests of intentionally defected elements show that failure of an oxide rod probably would not force immediate shutdown of a reactor, and that fission gases are released by a failed element in sufficient quantity to detect the failure. Although there have been instances of oxide release into the coolant as a consequence of severe failures, the oxide remained substantially intact in most of the defected elements.

Development Program. Most of the oxide fuel development and irradiation testing in this country to date has been related to enriched fuel reactor programs and has therefore been directed toward the objective of increased burnup. However, while much higher burnup is sought in enriched reactors as a means of reducing overall fuel costs, low unit fabrication cost is one of the major objectives in efforts to reduce fuel costs for natural uranium oxide.

The fabrication cost of the close-tolerance, sintered-pellet elements that are being used in enriched reactors today is not low enough to make the power cost for a natural uranium reactor competitive with that of a fossil-fueled plant. The cost of sintered pellet elements can be reduced by tolerance relaxation, especially in the lower-flux, slightly enriched reactors. However, mechanical compaction by swaging is being developed by a number of sites (including Hanford, Chalk River, and Savannah River) as an alternative and potentially cheaper route to suitable oxide elements for natural uranium reactors.

Both sintered and fused oxide are being used as starting materials for swaged elements. The fused oxide has been shown by SRL to swage to a higher bulk density than the sintered material. Higher density is advantageous from nearly all aspects: improved reactor physics, improved fission gas retention, increased thermal conductivity, and reduced mass migration. Bulk densities as high as 89 to 92% of theoretical have been achieved by swaging crushed and sized forms of sintered oxide and fused oxide, respectively. A density of 90% of theoretical is currently regarded as the minimum that is acceptable. Additonal data are needed on the performance limitations of swaged elements as a function of bulk density.

Swaged rods are performing well in irradiation tests in the VBWR, MTR, ETR, and at Hanford; rods of crushed oxide swaged in Zircaloy will be irradiated in quantities in the first core loading in PRTR. The Canadians expect to irradiate swaged rods in the NPD-2 if their appraisals indicate that these elements will be cheaper than sintered pellets. The Savannah River program is emphasizing swaged tubes, which are more massive and may, therefore, prove to be cheaper than swaged rods. In-pile tests of tubular specimens are under way at Savannah River. These specimens are clad with stainless steel, which is being used as a temporary stand-in for Zircaloy.

A major objective of the current Savannah River program for oxide is to supply a load of swaged tubes in time for startup of the HWCTR (third quarter-1961). Accomplishment of the program objective will require the successful adaptation of the swaging process to long Zircaloy-clad elements, refinements of end-sealing techniques, and demonstration of successful performance of Zircaloy-clad elements in a Savannah River reactor.

Other fabrication routes that have been investigated for compacting bulk oxide in Zircaloy include high-temperature isostatic compaction, explosive compaction, vibratory compaction, hot extrusion, and rolling. The most promising of these, vibratory compaction, is being used by GE at Hanford as a preparatory step for swaging to reduce the number of passes required and the consequent detrimental working of the cladding. Densities as high as 90% of theoretical were achieved in some instances in experiments with rods, and it appears that vibratory compaction alone (i.e., no subsequent swaging) might be adequate as a fabrication route for oxide rods. Further development of this process is under way at Hanford and Savannah River. At Hanford, vibratory-compacted elements are now being irradiated. In preliminary experiments at Savannah River, densities of 87% have been achieved by vibratory compaction of tubular shapes.

5.2.2 Uranium Metal

The fabrication development effort on metallic uranium has been concentrated on a tubular fuel element (2.06 in. OD, 1.47 in. ID, and 10 ft long) that is clad with Zircaloy. Nuclear Metals, Inc., (NMI) has successfully adapted a coextrusion process to the fabrication of such tubes. More than 50 tubes with cores of either unalloyed uranium or U-2 w/o Zr have been prepared by this process; most of the fabrication experience is with the latter material. The fundamentals of the process are considered to have been adequately demonstrated, although additional development along lines of production yield and quality improvement will be required.

Experience with detection of fuel failures in water-cooled reactors indicates that a failure of a metallic element in a full scale reactor can be detected when 2 to 5 grams of uranium have corroded. Although uranium will react relatively rapidly with D_2O at operating temperatures, laboratory corrosion data obtained at NMI on purposely defected Zircaloy-clad tubes of U-2 w/o Zr show that activity monitors should be capable of detecting a failure at an early enough stage to prevent gross contamination of the reactor complex.[19] The corrosion is characterized by an induction period of several hours duration. After 20 to 30 grams have corroded, the corrosion rate increases sharply. In the NMI tests, which were performed with unirradiated material, the corrosion rate of unalloyed uranium was higher by a factor of three than that of the U-2 w/o Zr; however, it is expected that a failed element of either type could be removed from a reactor during the induction period while the corrosion rate is relatively low.

The only in-pile data on the progress of metal failures at power reactor conditions are those obtained by the Canadians on a defected rod of U-2 w/o Zr which was irradiated in a loop of the NRX.[20] In this test, the monitors detected a burst of activity in the coolant after only 100 to 200 mg of uranium had corroded. The reactor was shut down within minutes of the detection of the burst, and the loop was cooled. No further significant corrosion took place in the 4 hr interval between shutdown and removal of the specimen from the loop. (It should be noted that this immediate shutdown would be required in a power reactor also.)

To be considered acceptable for a D_2O-moderated power reactor, metallic elements must be capable of withstanding an average exposure of at least 3000 MW-d/metric ton-U at a maximum metal temperature of about 1000°F and at an external pressure of about 1000 psi. Because no facility is available for irradiation tests at these conditions, directly applicable data are virtually nonexistent. It is not known, therefore, whether the desired exposure can be achieved without excessive failures. Present data consist of the results of low temperature, low pressure irradiations at Savannah River, a single test at higher temperature and pressure but low burnup in the VBWR, and tests in the NRU E-20 loop.

At Savannah River, a total of 13 coextruded tubes of uranium with zirconium-base cladding have now been irradiated; exposures in the range of interest for power reactors were achieved in some of the tests.[2,3] Eight of the tubes were of U-2 w/o Zr, and the remainder were of unalloyed uranium. In general, the irradiation conditions for the U-2 w/o Zr were the more severe, and three of these tubes failed during irradiation. The Savannah River data are useful in preliminary comparisons of alternative fuel compositions, but the test conditions were not representative of power reactor operation and may have been unfavorable with respect to effects of cladding temperature and coolant pressure. In the VBWR test, a tube of U-2 w/o Zr clad with Zircaloy-2 has reached an average burnup of 1200 MW-d/metric ton-U at an external pressure of 1000 psi, a cladding temperature of 575°F, and a maximum metal temperature of 800°F.[4] The maximum cladding strain in this tube is about 0.7%. A tube of unalloyed U was irradiated at high temperature and pressure in the NRU E-20 loop to a burnup of 950 MW-d/metric ton-U (max). In Canadian irradiations of Zircaloy-clad rods of U-3.9 w/o Si at power reactor temperatures and pressures, failures did not occur until cladding strains reached about 2.5%.[5]

The HWCTR is being constructed to fill the need for an experimental facility in which fuel irradiations under power reactor conditions can be conducted in quantity. This reactor, which is described in Section 7.2 of the Appendix, is scheduled to start up in the third quarter of 1961. It is anticipated that the first post-irradiation data for the initial load of test fuel elements will be available in early 1962. In the meantime, the irradiation test in the VBWR will be resumed and four specimens (U-2 w/o Zr, unalloyed U, U-1 w/o Si, and U-1½ w/o Mo) will be irradiated.

A program has also been initiated to increase the strain that can be accommodated by the cladding. Included in this program is the substitution of Zircaloy-4 for Zircaloy-2, the objective being to decrease the hydrogen pickup by the cladding during reactor operation.

5.3 HEAT TRANSFER AND FLUID FLOW

5.3.1 Current Status of Technology

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.

Greater uncertainties exist in the heat transfer and fluid flow calculations for a boiling reactor than for a nonboiling reactor, and experimental investigations are required to determine the pertinent characteristics of the fuel assemblies that must be employed.

The best information available on the burnout heat flux in forced convection boiling is that reported by Westinghouse for flow in short tubes and rectangular channels.[37] The Westinghouse data emphasize operation at 2000 psi, and there is a dearth of information in the 700 to 1000 psi pressure range; this range is of particular interest in the design of boiling D_2O power reactors.

The existing data are inadequate from another standpoint, namely, the simplicity of the test sections that were employed. A typical fuel assembly for a boiling reactor consists of several concentric tubes or a bundle of rods. It is not known to what extent the heat transfer in such assemblies can be represented by tests with simpler geometry. The distribution of the boiling coolant is more difficult to define than in a single coolant channel, particularly in rod bundles where the coolant can flow laterally between regions of different hydraulic characteristics and heat generation.

Methods are available for predicting the head-loss characteristics for boiling flow, and thereby identifying regions of flow instability.[39,40] Calculations indicate that the instabilities in a boiling reactor occur at lower pressures and at higher inlet subcoolings than are of interest for D_2O-moderated power reactors. Head-loss data are needed for a wide range of powers at geometrical configurations and pressure levels that are of direct interest. Laboratory studies are required to determine the mode of flow and the ratio of the velocities of the two phases in order to obtain better models for calculations beyond or within the range of the experimental data. This latter information will also be useful in kinetic analyses of boiling reactors, since the distribution of void fractions affects the void coefficient of reactivity.

Vibration of fuel assemblies in a boiling reactor should not be a major problem. Investigation of vibration phenomena are being made. Although excessive vibrations resulting from flow fluctuations have been experienced occasionally in boiling experiments, these instances have always been in highly subcooled flow. This question can be explored in loop tests of fuel assembly mockups. However, a faithful mockup of a boiling assembly will be difficult to construct, and the results of mockup tests must be accepted with reservations. In-pile tests are necessary for a better appraisal of the mechanical stability and integrity of candidate fuel assemblies for a boiling reactor.

5.3.2 Research and Development

The Columbia University boiling heat transfer loop has been modified for high pressure operation with steam generation. The new loop has a rated pressure of about 1100 psig, an electrical heat generation capacity of about 3500 kw direct current, and can accommodate certain test sections up to about 10 ft in length. Preliminary tests will be run with a single-heated rod and an annular flow passage. These will be followed by tests with rod bundles. The tests will include variations in test section length, flow rate, pressure, and rod spacing methods. With the assistance of NDA, tests will be made to determine burnout heat fluxes, coolant void distribution, and pressure drop.

A high pressure test loop for studying two-phase flow with heat generation has been constructed at SRL. The loop has a rated pressure of 1500 psig and an electrical heat generation capacity of 300 kw direct current. Test sections up to about 18 ft in length can be accommodated. Tests will include measurements of pressure drop and void fraction with boiling inside single tubes, and observations of possible flow and pressure fluctuations with two or more tubes operating in parallel. This experiment is expected to be completed by the end of 1960.

A high pressure loop for cooling a fuel element with boiling D_2O coolant will be installed in the HWCTR. This loop will permit tests of the mechanical performance of the fuel elements and will give information on corrosion and erosion and on possible flow instabilities with clustered rod elements. The fuel elements will be about 10 ft long.

An investigation of two-phase flow for clustered rod elements would be extremely difficult in a high pressure loop with electrical heat generation. Consequently a low pressure cold flow test loop is planned for this study. The loop will accommodate mockups of full length elements for the large scale reactors (i.e., up to about 20 ft in length) and will simulate the boiling process by means of gas injection. The flow distribution within the element, pressure drop, and void fraction will be investigated. Transparent test section walls can be used for the tests and high speed photographs of the two-phase flow will be taken.

None of the facilities for testing clustered rod elements can accommodate test sections long enough to simulate the full scale reactor. Tests of full length elements would be of interest in connection with proof-testing the final fuel element designs. These would include observations of possible flow-induced vibrations and flow fluctuations within the element.

SRL is developing digital computer methods for analyzing two-phase flow in reactors with subdivided fuel elements. The methods will use the results obtained in the various test loops.

5.4 COMPONENTS AND AUXILIARY SYSTEMS

5.4.1 Pressure Tubes

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 has adequate mechanical properties and corrosion resistance and 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. 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, CVTR, and NPD-2 reactors, all of which will employ pressure tubes of Zircaloy-2.

Data on the effect of irradiation on the mechanical properties of Zircaloy indicate that for annealed or moderately cold-worked material there is no appreciable reduction in tensile or yield strength as measured during post-irradiation testing. At high neutron exposure, there is a significant reduction in uniform elongation values (e.g., yield strength at 0.2% offset coincides with ultimate tensile strength), although the area-reduction property is only slightly affected.

If the Zircaloy is very heavily cold worked prior to irradiation, the tensile and yield strengths may be reduced by in-pile annealing effects. In instances where this phenomenon has been observed, the strengths still have been significantly higher than those of irradiated Zircaloy in either annealed or moderately cold-worked state.

The long-term creep properties of Zircaloy are not well known. In-pile tests for determining creep strengths are only now under way, and it is too early to say whether a high neutron exposure will affect this strength property.

The basic corrosion resistance of Zircaloy-2 is sufficiently high that under normal corrosion behavior at 570°F no more than 0.1 mil would be corroded in 10 yr. If the tube surface is contaminated, the corrosion rate may increase severalfold; even this higher rate would be considered small when expressed in terms of penetration. Of perhaps greater concern is the absorption of hydrogen (or deuterium) produced by radiolytic decomposition of water. When the hydrogen exceeds its solubility limit, it precipitates in the form of zirconium hydride and ultimately degrades the mechanical properties. It is known that elimination of nickel from Zircaloy-2 results in less hydrogen pickup during corrosion in water or steam, and it is for this reason alone that Zircaloy-4 is receiving increased attention.

Most of the development work to date on Zircaloy pressure tubes has been 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 PRTR, only a few had minor defects and even these will be installed 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 Hanford 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 pressures of 900 to 1500 psi. One of these tubes was recently sectioned for examination after irradiation for about 2 yr. The results of the examinations to date are reported to be generally satisfactory except that one section of the tube deteriorated after inadvertent exposure to conditions that are extraneous to the power reactor program. A section that had been irradiated at the edge of the peak flux area exhibited no recrystallization or inclusions. The examinations are continuing but there was no obvious change in tube dimensions, and no evidence of localized corrosion.

At Chalk River, a 5-in. diameter Zircaloy-2 pressure tube has been in service in the NRX reactor for 3 yr 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.

Under the ECNG/FWCNG program the feasibility of fabricating a satisfactory welded seam Zircaloy tube has been demonstrated. Eighty lineal feet of 5-in. diameter, 0.120-in. wall tubing has been fabricated and tested.

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 these 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 ECNG-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.

5.4.2 Pressure Tube Zr-SS Joints

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 metallurgical 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, a 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 NMI in the development of bonded joints, and specimens of tubular joints of practical size are being evaluated.[11] In a burst test, one specimen of a bonded joint (1.9 in. OD $\times$ 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.

Combustion Engineering is preparing a test joint for CNEC, employing a nickel-iron transition section. Zircaloy will be rolled and brazed to the nickel-iron section, the two having similar coefficients of expansion, and the transition piece will be welded to the stainless steel tube.

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

5.4.3 D_2O System High Pressure Seals, Joints, and Closures

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 the temperatures and pressures of interest. The economic import of D_2O losses is shown in Fig. 5.1, which relates the loss rate to power costs for reactors 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.[10] AECL and GE have investigated component leakage. These programs have provided data on leakage rates through static joints and closures of conventional design, and on valve stems, pump seals, turbine seals, and tube fittings. The programs in progress include all of the out-of-pile tests deemed necessary at the present time.

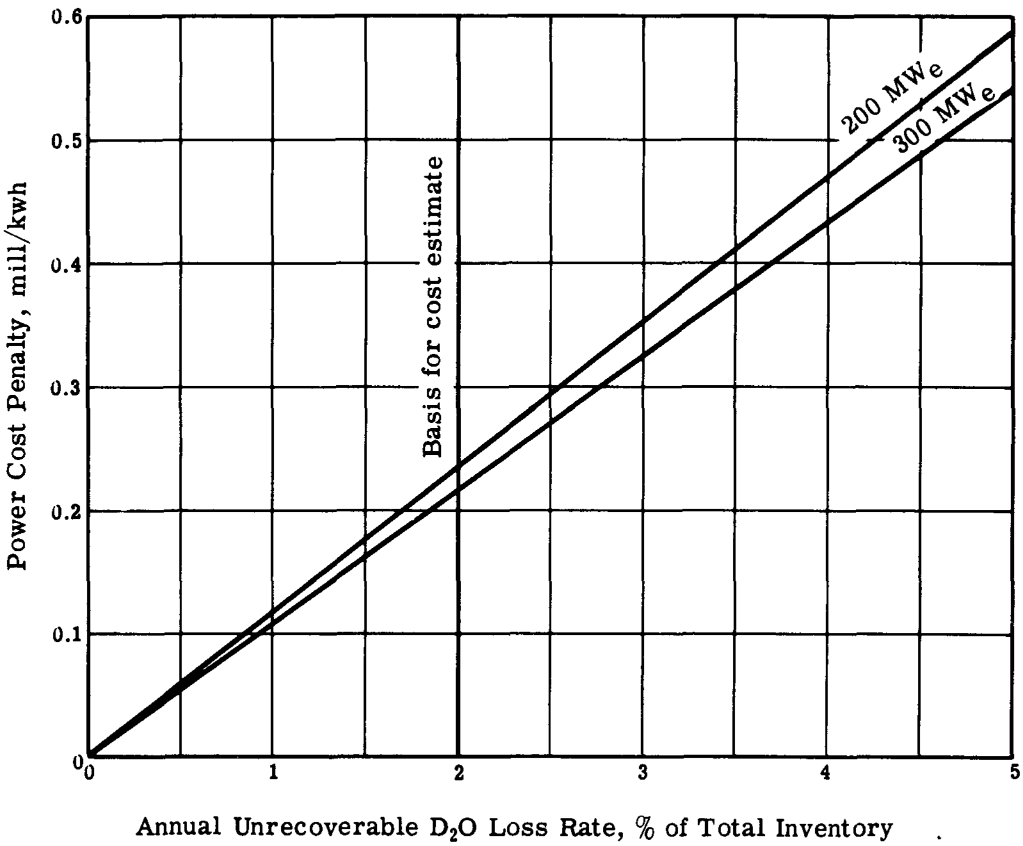


Fig. 5.1 — Effect of D_2O loss rate on power cost for oxide-fueled boiling D_2O, direct cycle plants

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. Information of the latter type will begin to accumulate later this year when the PRTR is placed in operation.

The status of the leakage investigation programs for each major category of equipment is summarized below. There are indications that the D_2O loss rate assumed in the economic studies should be reduced.

Valve Stems

An exploratory investigation of stem leakage from conventional globe and gate valves has been completed at SRL. In this program, valves ranging up to 6 in. in size were evaluated in static and dynamic tests with water at 500°F and 1000 psi. Detailed results of these tests may be found in SL-1815. For example, during a 100 cycle test, the average water leakages from a 3 in. globe valve and a 6 in. gate valve were 0.09 and 0.6 lb/yr, respectively. Leakage rates of this order of magnitude can be considered as insignificant.

AECL is undertaking a leakage measurement program similar to the one described above. The first phase of this program is the measurement of leakage from valves (maximum size: 6 in.) in a recently installed flow loop at the Manby Station of the Ontario Hydroelectric Power Commission in Toronto. This loop is designed for operation at a maximum temperature and pressure of 600°F and 2000 psi.

Pump Seals

The successful operation of the mechanical shaft seals of D_2O pumps requires the flow of a small amount of fluid past the sealing faces. Most of this flow is recovered as liquid in a gland which is incorporated in the seal assembly, but a small fraction of the flow escapes as vapor. In substantially all instances in which seal leakage has been investigated by various manufacturers and systems operators, only the recoverable liquid losses have been measured. The only available data on the vapor losses are those obtained in preliminary measurements on a pump in a high temperature flow loop at SRL. Although reliable quantitative measurements have not yet been made on this pump, initial indications are that the vapor loss will be acceptably low.

Apparatus was constructed for use in an investigation at SRL of both liquid and vapor leakage through shaft seals. Leakage through seal assemblies is being measured in this apparatus, which simulates an operating pump in regard to temperature, pressure, and static forces, but does not simulate dynamic forces. The variables being investigated include shaft size ($4\frac{1}{2}$ in. max), water pressure, and shaft speed. "Start-stop" tests are included in the investigations. The target for completion of the present program is December 1960.

The results of the tests described above are being supplemented by measurements of leakage in vendor's tests of the HWCTR pumps, and by further monitors of pumps in test loops at SRL, Hanford, and in two Canadian installations (Peterborough and Manby).

Turbine Seals

The turbine-generator is a source of D_2O loss that is unique to a direct cycle reactor plant. There are no quantitative data available now on the leakage through turbine seals. However, large measurements have been made during operation of the 5 MW_e turbine-generator of the EBWR. The EBWR was originally designed to use D_2O and was therefore equipped with seals that were designed for low leakage rates and for leakage recovery. Although this unit is much smaller than turbines of a full scale power reactor, a quantitative indication of the leakage has been obtained. The results of two series of tests, both with and without the vapor recovery systems, show that the EBWR turbine seals perform better than specified by design criteria. The D_2O loss rate was approximately 0.65 lb/month. This amount of leakage can certainly be tolerated with no significant effect on power cost.

The overall economics of the boiling D_2O reactor are not strongly dependent upon the outcome of these tests, because the problem of turbine leakage can be circumvented by resorting to an indirect-cycle plant. Preliminary appraisals indicate that substitution of the indirect cycle will result in power costs about equal to those of the direct cycle plant and will reduce substantially the possibility of external system D_2O loss.

Static Joints and Closures

Conventional flanged joints and closures that incorporate stainless steel gaskets with asbestos filler have been tested for leakage at SRL, in connection with the program of component evaluation for the HWCTR. Conventional tubing connectors also have been included in this program. The SRL tests consist of cyclic operation of the components to simulate startup and shutdown of a reactor. Tests at peak conditions of 1000 psi and 500°F are complete, and similar tests at 1500 psi are in progress. The results of the tests indicate that although further testing of specific designs will be necessary, leakage from conventional joints and tubing fittings will not be a major problem if the components are rigorously inspected and carefully assembled. The durability of seals which must be disturbed during refueling operations remains to be ascertained.

5.5 STABILITY AND SAFETY OF BOILING REACTORS

Reactors that are cooled and moderated by heavy water are relatively slow in responding to disturbances and are easily controlled. However, some designs have a positive void coefficient of reactivity. The existence of this characteristic in a boiling reactor has raised questions as to (1) whether positive reactivity feedback through the void coefficient can lead to an uncontrolled 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 heat transfer burnout. Present indications are that the existence of a positive void coefficient does not affect the control feasibility of a boiling reactor that is fueled with uranium oxide. The control situation has not been resolved yet for metal-fueled reactors. The negative temperature coefficient for the metal fuel is much smaller than that for the oxide fuel and therefore does not exert as large a restraint on the positive void component.

Experimental data on the total coolant worth of clustered rods of uranium oxide and uranium metal with D_2O moderator and coolant have been obtained at SRL, Hanford, and Saclay, France. No data have been obtained for partial removal of coolant, as in a boiling reactor. Reasonable agreement was obtained when the experimental data were compared with results of semi-empirical calculation methods developed in France and Sweden.[36] The comparison showed no clearcut choice in the calculation procedure to be used for the boiling D_2O, oxide-fueled reactor lattice, which lies outside of the range of the experiments. The predicted values of the void coefficient of reactivity for this reactor at design power, as obtained from the French and Swedish methods, are $+7.5 \times 10^{-5}$ and $+3.8 \times 10^{-5}$ $\Delta k_{eff}/k_{eff}$ per percentage point increase in vapor volume fraction, respectively. The higher predicted value obtained from the French method was used in reactor transient calculations.

The positive power coefficient of reactivity resulting from void formation is of more direct interest than the void coefficient of reactivity in considerations of reactor stability. Detailed calculations for an oxide-fueled boiling reactor show that reactor power has small effect on the average volume fraction of coolant vapor in the design power range. Also at design power, a void coefficient of $+7.5 \times 10^{-5}$ $\Delta k_{eff}/k_{eff}$ per percentage point increase in void fraction results in a power coefficient of reactivity of $+1.5 \times 10^{-5}$ $\Delta k_{eff}/k_{eff}$ per percentage point increase in power. This positive coefficient is overshadowed by the large negative power coefficient associated with an increase in temperature of the oxide fuel.

A typical example of the response of an oxide-fueled, boiling D_2O reactor (the S&L-NDA design) to a step insertion of reactivity is shown in Fig. 5.2. It will be noted that the power excursion is self-controlled, i.e., power increases rapidly and then falls because the negative reactivity effect accompanying the increase in fuel temperature overrides the positive effect of an increase in steam production. These transients were calculated under the assumptions that the vapor distribution in the reactor at any time is that given by steady state relationships, and that coolant flow, inlet temperature, and pressure are constant during the time of the transient. The results, applicable for either a direct or indirect steam cycle, show that an uncontrolled excursion does not result. It is seen by reference to Fig. 5.3, which shows the power corresponding to heat transfer burnout at a given flow rate, that the increase in power following a reactivity insertion of +0.0003 k (which is about 10 cents at 7500 MW-d/metric ton-U) is less than that required to produce burnout. A reactivity insertion of this magnitude is roughly equivalent to that introduced by a fuel element falling into a central lattice position. The burnout line in Fig. 5.3 represents a correlation developed by Westinghouse.

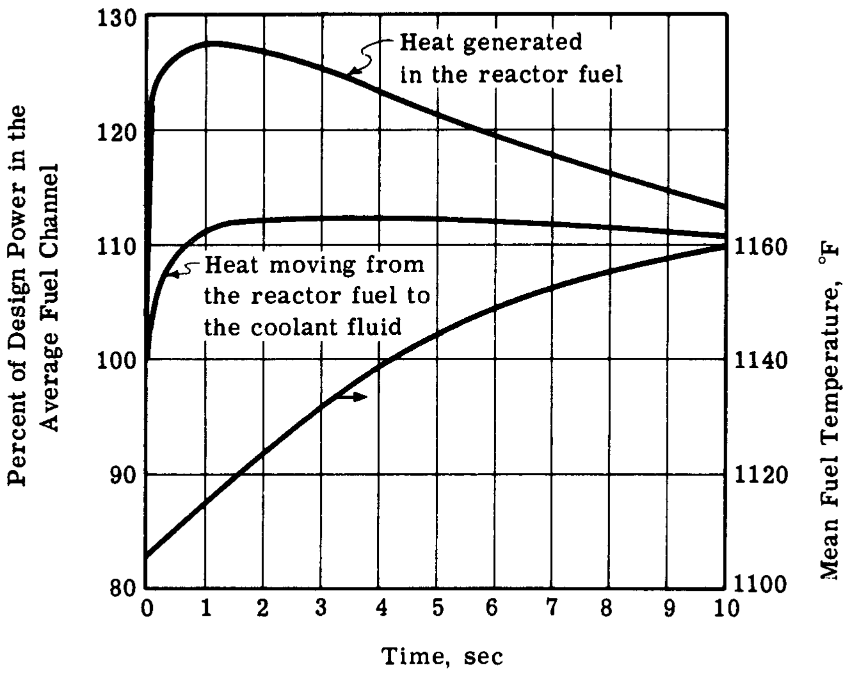


Fig. 5.2 — Typical response of boiling UO_2-fueled reactor to step increase in reactivity. The step change in reactivity is 3×10^{-4}, added at zero seconds. The coolant void coefficient of reactivity is $+7.5 \times 10^{-5}$ $\Delta k_{eff}/k_{eff}$ per percentage point increase in vapor volume fraction.

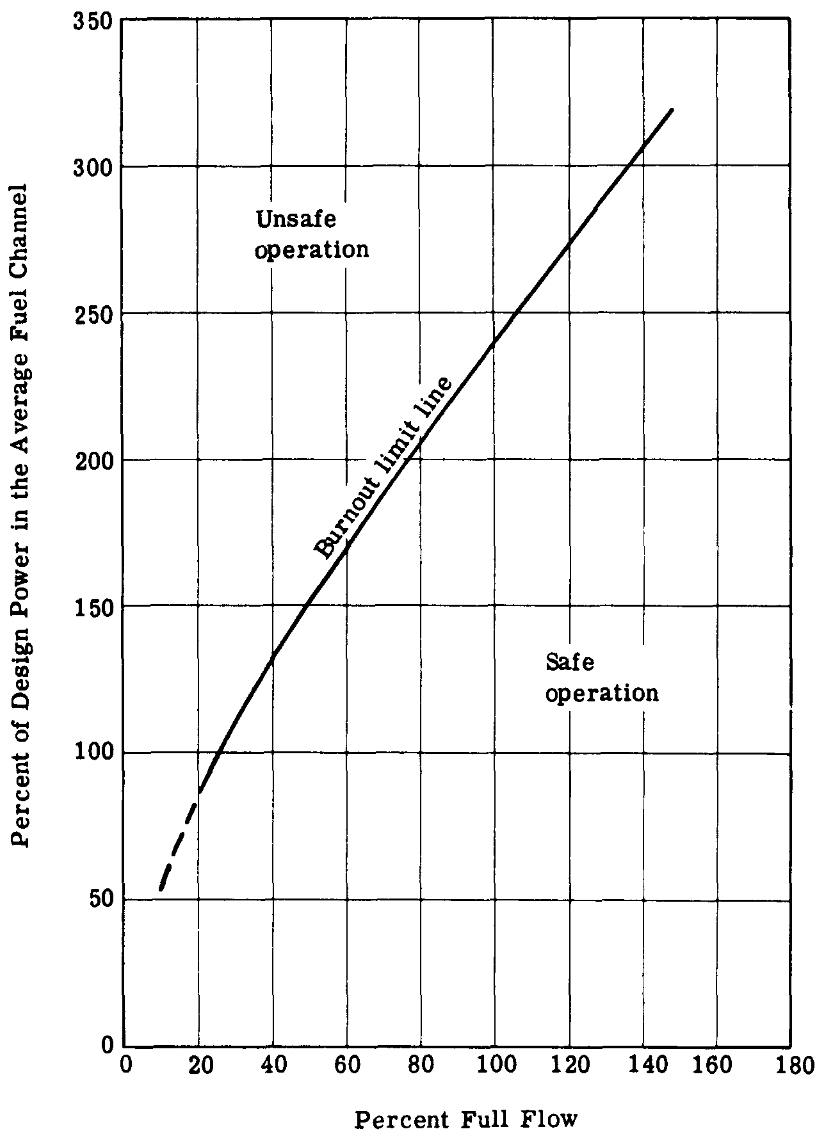


Fig. 5.3 — Burnout limits for fuel assemblies of oxide rods cooled by boiling D_2O

100% design power = 16.8×10^6 Btu/hr; 100% design flow = 15.7×10^4 lb/hr; axial cosine heat flux distribution.

Burnout limits are based on the following Westinghouse correlation (Reference WAPD-188):

$$\frac{\phi_{BO}}{10^6} = 0.182\left(\frac{H_{BO}}{10^3}\right)^{-2.5}\left(1+\frac{G_t}{10^7}\right)^2 e^{-0.0012L/D}$$

where ϕ_{BO} = burnout heat flux, Btu/hr-ft^2
H_{BO} = coolant enthalpy at burnout, Btu/lb
G_t = channel mass flow rate, lb/hr-ft^2
L = heated length from inlet to burnout point, ft
D = equivalent diameter, ft

The results in the example reported above are dependent upon the magnitudes of the void and temperature coefficients, as well as upon the assumed behavior of the two-phase coolant in the reactor during a transient. The results of calculations in which the void and temperature coefficients were varied over a reasonable range indicate that if the void coefficient were doubled, or the temperature coefficient halved, the peak power attained after a step insertion of reactivity would not increase significantly.

There are a number of detailed design problems that have been recognized but not yet thoroughly investigated. In the start-up range, the change in vapor volume fraction, and therefore in reactivity, associated with a unit change in power is greater than that in the operating range. Off-design conditions will be investigated prior to specifying the control system.

Pressure drop and flow characteristics of the boiling D_2O fuel channels have been analyzed, and it has been concluded that flow oscillations that could interact detrimentally with the reactor dynamics are unlikely.

An additional aspect of the positive void coefficient in a pressure tube reactor is the possible self-propagation of local increases in vapor volume. If an increase in vapor fraction in one channel (caused, for example, by a local flow reduction) were accompanied by an increase in local reactivity and power, both the vapor fraction and the reactivity would increase further. This situation has been examined for an oxide-fueled reactor, and it is concluded that only a small local power rise ensues before equilibrium is again established.

The metal-fueled boiling reactor differs in its behavior from the oxide-fueled reactor primarily in the dynamic characteristics of the fuel itself. The fuel temperature power coefficient of reactivity is smaller, and the thermal diffusivity of the fuel element is much larger. Furthermore, the void coefficient itself may be larger. The calculated result, to date, indicates that the metal-fueled boiling reactor will not be stable. If this is confirmed by experiment, it may be necessary to limit the use of metal fuel to nonboiling systems.

Techniques for calculating the transient response of the boiling, pressure tube reactor will be developed. The effects of the external systems on the transient behavior of the reactor will be included in the analysis. The methods will be used to investigate the response of the reactor to changes in load demand and reactivity. Some analysis of the prototype selected from the results of the physics program will also be conducted and a preliminary hazards report will be written. Tests on nuclear safety, to be performed in the HWCTR, will be a source of additional data on the characteristics of D_2O-moderated reactors operating at high pressure and temperature conditions.

5.6 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. However, some caution must be exercised since erosion, corrosion, etc., may be a problem with hot D_2O. Several organizations including du Pont, GE, and S&L are investigating the coolant chemistry required with less expensive materials in the primary system. This work may result in considerable plant cost savings.

5.7 FACILITIES

5.7.1 Test Facilities Available or Under Construction

The long range development of D_2O-moderated power reactors requires both reactor facilities for physics experiments and in-pile testing and engineering laboratory facilities for hydraulic and heat transfer experiments and development of reactor materials, fabrication processes and components. The major contributions to the out-of-pile work are being made by the Savannah River Laboratory of the USAEC, and the Chalk River and Toronto facilities of AECL. These organizations are supplementing their own efforts and facilities by contract with industrial organizations, particularly in the development of fuel element and pressure tube materials and fabrication processes. In addition, the Columbia University Engineering Research Laboratory, with assistance from NDA, is carrying on an experimental program in the Task X loop system to determine the heat transfer limits and possible vibration phenomena for boiling D_2O reactors with various rod bundle and concentric tube fuel element configurations. This work will complement flow test work in the SRL multiple channel parallel flow test loop, the large flow test loop at AECL-Toronto, the NRU E-20 loop, and the bayonet loops in HWCTR.

These organizations and others such as Hanford and Carolinas-Virginia Nuclear Power Associates are also contributing to D_2O power reactor technology with shorter term development programs in support of specific designs of D_2O test, prototype, or demonstration power reactors which are either under construction or scheduled for construction in the relatively near future. These reactors include PRTR, CVTR, HWCTR, FWCNG, NPD-2, and CANDU. None of these reactors are of the boiling D_2O type which is the current objective of the long-term development program in the United States, but all are expected to be important sources of operating and in-pile performance data needed for this program.

The following critical and exponential facilities will be used for experimental physics work.

Process Development Pile

The Process Development Pile (PDP) at SRL has a tank 16 ft in diameter and 15 ft high. It has been converted into a flexible critical facility to obtain accurate buckling measurements at room temperature for a number of heavy water lattices.

Pressurized Sub-Critical Experiment

The Pressurized Sub-Critical Experiment (PSE) at SRL is a source-fed, heavy water exponential experiment that can operate at temperatures up to 215°C and pressures of about 300 psi. It will be used primarily to measure temperature and void coefficients of reactivity.

The Pawling Lattice Test Rig

The Pawling Lattice Test Rig (PLATR) is a small critical facility designed to permit rapid and accurate measurements of k_∞ coolant void coefficients, and of the effect on reactivity of engineering changes to the lattice.

The following reactor facilities will be used for experimental work or relied upon as sources of important performance data in the long-term development program. There are in addition to CVTR, PRTR, NPD-2, FWCNG, HWCTR, and the D_2O prototype power reactors, more detailed descriptions of which can be found in the Appendix, Section 7.

EBWR – Experimental Boiling Water Reactor, a 5 MW_e, light water, power reactor experiment now in operation at the Argonne National Laboratory. The EBWR was originally designed to use boiling D_2O in a direct cycle and therefore the 5 MW_e turbine is equipped with seals that were designed for low leakage rates and leakage recovery. Although this is a small unit, it will give some indication of leakage loss rate to be expected from turbine shaft seals. This plant will also provide data on steam separation and possible carry-over of fission products to turbine equipment.

ETR – Engineering Test Reactor at the National Reactor Testing Station at Arco, Idaho. This reactor is one of the reactor facilities being used for in-pile radiation testing of UO_2 fuel elements being developed by GE.

MTR – Materials Testing Reactor at the National Reactor Testing Station at Arco, Idaho. This is one of the reactor facilities being used for in-pile radiation testing of UO_2 fuel elements being developed by GE.

NRU – A heavy water test reactor at the Chalk River Laboratory of AECL. This reactor has a high pressure loop facility which will be used for in-pile irradiation testing of long-length fuel elements.

SRP – The Savannah River Plant production reactors provide facilities for irradiation testing of long length fuel elements.

VBWR – Vallecitos Boiling Water Reactor, a 5 MW_e boiling light water power demonstration reactor built by the General Electric Company at their Vallecitos (Cal.) Laboratory. This reactor is being used for irradiation testing of fuel elements under power reactor operating conditions.

5.7.2 Additional Test Facilities Required

None of the heat transfer and hydraulic test loop equipment available at SRL, Columbia University, and AECL facilities will exactly duplicate the combined parameters of heat generation: length, diameter, fuel configuration or coupling arrangement between the pressure tube and the steam drums. If it is found necessary to obtain these data prior to construction of a prototype reactor, a modification to an existing facility or a new facility may be required.

5.8 PROTOTYPE OR DEMONSTRATION REACTORS

Within this category of heavy water reactors are the CVTR and NPD-2 which are under construction and have been noted in the section above in connection with their utility as test facilities. In addition, there is the ECNG-FWCNG gas-cooled, D_2O-moderated reactor, which is discussed both in Section 7 herein and in the 10-year program report on gas-cooled reactors, and CANDU, which is a 200 MW_e heavy water power reactor to be built in Ontario, Canada by AECL.

The accumulation of data from the PRTR, HWCTR, and CVTR will add emphasis to the need for a natural uranium-fueled prototype reactor. At this stage of the program the design specifications for the prototype need not be made firm, but for purposes of program planning it can be assumed that the S&L-NDA 70 MW_e prototype design[13,14] is representative. Plant characteristics for the prototype reactor are presented in Table 7.1.

A full-scale demonstration reactor would be included in the long-range program. For planning purposes, it has been assumed that this is a 300 MW_e, boiling D_2O-cooled, direct cycle plant as described in Section 7.1.

5.9 DESIGN AND EVALUATION STUDIES

The performance potential of heavy water, natural uranium power reactors should be re-evaluated periodically to factor in not only those technical developments which are accomplisned within this heavy water reactor program but also pertinent technical developments from other programs and activities. The program includes an up-dating of potential improvements to keep the configuration and economics of the full scale reference design power reactor current with technical developments. The overall program will be correspondingly modified in scope, emphasis, and direction as necessary. The items to be included initially in the evaluation are discussed below.

Dispersion-Hardened Uranium. This essentially metallic fuel, sometimes referred to as sintered uranium powder (SUP), is being developed at a modest level of effort at NDA and jointly by Massachusetts Institute of Technology and NMI. The dispersion of UO_2 throughout metallic uranium may prevent intergranular slip which is the mechanism which permits growth during irradiation. If the process can be extended to alloys of uranium, the water corrosion problem may be reduced.

Plutonium Recycle. The program at Hanford being conducted by GE will start producing operating data on the PRTR in 1961. Also fuel element fabrication data with recycled plutonium will be forthcoming when the best method of using the concept is defined.

Heavy water-moderated reactors, having a high initial conversion ratio, produce more plutonium than most other reactor types. The economic advantages of plutonium recycle should therefore be greater for this reactor type.

The main question to be answered, prior to determining the place of plutonium recycle in the program, is fuel cycle cost of irradiated plutonium.

Steam Separation. In the current D_2O reactor designs, many tons of heavy water are held up in the steam drums. This represents a large capital investment and operating expense which does not exist for light water-cooled reactors.

In early work on the program, it became apparent that very little effort has been put into reducing the water holdup and that work which has been done in industry is proprietary in most cases. Argonne National Laboratory is doing some development work along these lines in connection with the EBWR program. The du Pont Company is investigating fabrication methods of smaller, more efficient heat exchangers and steam generators.

The development of steam drums, in-line separators, and other devices which might be used to reduce D_2O inventory and reactor plant size should be followed closely and supplemented by development programs as required.

H_2O Fog Cooling – Direct Cycle. The inventory of heavy water in the cooling circuit of a 300 MW_e direct cycle plant is about 360,000 lb and represents an investment of \$10,000,000. At the rate of 12.5% for nondepreciating capital, this represents a charge of \$1,250,000/yr plus about \$200,000/yr for D_2O losses (2%/yr). Elimination of D_2O from the coolant circuit would reduce power cost by 0.6 mill/kwh in D_2O charges alone. By substituting H_2O fog, additional savings would accrue from the use of conventional steam equipment and the better thermodynamic properties of H_2O. Other factors which could affect the power cost are the poisoning effect of H_2O and the increase in reactivity from the use of SUP or other metallic fuel.

Investigations of H_2O fog cooling for H_2O-moderated reactors are underway in this country at NDA in a cooperative program with CISE and Ansaldo of Italy, under joint USAEC-Euratom research and development program.

H_2O Fog Cooling with Nuclear Superheat. A logical extension to fog cooling is the addition of superheating in the coolant channel. Unfortunately, there is no satisfactory high temperature fuel cladding material which is sufficiently transparent to neutrons. Should a beryllium alloy, or other low cross section, high temperature material, be developed which can resist dry steam corrosion, nuclear superheating in a fog-cooled reactor will be quite attractive.

Boiling D_2O with Nuclear Superheat. A reactor which incorporates both boiling and superheating is very desirable. The pressure tube type of core is especially suited to this type of orientation, particularly if reasonably efficient in-line steam separators are developed. The full potential of superheat is dependent upon the successful development of a low cross section, high temperature, cladding material such as a beryllium alloy.

Boiling D_2O Indirect vs Direct Cycle. A recent study by NDA and Sargent & Lundy shows that the direct and indirect cycles with boiling D_2O coolant are essentially equal in predicted power cost. The indirect cycle reduces the possibility of D_2O loss from the external system and H_2O in-leakage and makes unnecessary the development of special turbine and condenser seals.

Pressurized D_2O vs Boiling D_2O Cooling. As indicated in Table 4.2, about a 1 mill/kwh penalty is incurred when pressurized D_2O coolant is substituted for boiling D_2O. Although there is no reason at this time to review the choice, developments in fuel element fabrication, reactivity prediction, and performance of the PRTR or CVTR may indicate the need for a re-evaluation of the two reactor types.

Gas Cooling. The heavy water-moderated, gas-cooled reactor is being developed by ECNG, FWCNG, and GNEC. As the program progresses it will be desirable to compare this concept with the other D_2O-moderated reactor types.

Organic Cooling. In the initial phase of the S&L-NDA program, during the selection of a reactor type, the organic-cooled version was eliminated because of its poor performance using Al or SS structural and clad materials. Development of satisfactory SAP or Be alloys would appreciably improve performance of the reactor. The Canadians are currently investigating this reactor concept under the auspices of AECL.

Insulated Pressure Tube vs Calandria. Two types of cores are being considered for power reactors in the D_2O program: internally insulated pressure tubes in a moderator tank (CVTR, FWCNG, and du Pont studies) and pressure tubes plus calandria tubes with gas space insulation (NPD-2, PRTR, and S&L-NDA studies). Both concepts have their advantages and disadvantages. Operation of the reactors now under construction will provide data from which a choice may be made for any particular reactor type.

Reactor Instrumentation. The pressure tube reactor is sufficiently different from pressure vessel reactors to make it worthwhile to evaluate the need for in-pile instrumentation. The use of natural uranium fuel increases a desire to know accurately the conditions in each fuel element in order to properly program refueling operations. Knowledge of coolant void fractions in each pressure tube might make it possible to detect incipient fuel element failure. The need or desire for new instrumentation should be coupled with a periodic determination of that instrumentation which has been developed, thus guiding this phase of the program.

On-Power vs Off-Power Refueling. The Canadian reactors (NPD-2 and CANDU) will use on-power refueling. The NDA-S&L studies indicate that burnups almost as high as the Canadian predictions can be obtained with multizone off-power refueling with axial inversion of elements if desired. Therefore, the choice would be made on the basis of the mechanical complexity and reliability of the refueling machine, reactor downtime, control requirements, reactor power patterns, control element poison effects, and fuel element reliability. Since control and refueling are closely coupled, a meaningful evaluation must wait until more detailed studies of heavy water-moderated control systems are made and the Canadian refueling machine is tested.

Up-Dating the Reactor Design and Economics. Periodic reviews of the potential full scale plants should be made as more complete experimental data are evolved. This should be done, at a minimum, just prior to starting the design of a reactor plant.

5.10 SUMMARY OF RESEARCH AND DEVELOPMENT PROGRAM

The current development program described above is considered to be adequate for the development of large heavy water-moderated reactors. Realization of the long-range potential is dependent upon the successful completion of several efforts which are being conducted in support of the overall power reactor program. These include:

UO_2 fuel element cost reduction
Alternate fuel development (metal, SUP, etc.)
Thorium-U^{233} fuel cycle
Plutonium fuel element fabrication
Beryllium alloy development
Improved reactor materials
Heat transfer and burnout experiments with H_2O fog and dry steam coolants.

6. COST REDUCTION POTENTIAL

It is anticipated that the power cost estimates for the 300 MW_e boiling D_2O pressure tube reactor plant, as reported in Section 4 for the current year, can be appreciably reduced for succeeding generation reactors. These potential cost reductions, based on 1960 costs to have comparative meaning, are divided into two categories:

1. cost reductions, as experience is gained in fabrication of components and operation of the power plant,
2. cost reduction by improvement of the cycle, fuel, and available materials resulting from the 10-year development program outlined in Section 5.

The type and estimated values of the potential cost reductions are discussed in the following paragraphs and are summarized in Tables 6.1 and 6.2. It is believed that the power costs for the 300 MW_e, boiling-D_2O direct cycle reactor could be reduced from 8.6 mills/kwh to 6.4 mills/kwh as indicated below:

	Power Cost, mills/kwh		
Cost Item	Current Technology	Successive Plant Improvements	Plant Potential
Capital investment	4.69	4.11	3.94
D_2O inventory	1.28	1.10	0.64
Fuel cycle	1.51	1.03	0.90
D_2O makeup	0.21	0.18	0.11
Operation and maintenance	0.58	0.58	0.58
Working capital	0.20	0.14	0.12
Insurance	0.13	0.13	0.13
Total	8.6	7.3	6.4

6.1 COST REDUCTION POTENTIAL IN SUCCEEDING GENERATION REACTORS

Once a reactor of a particular concept has been built and operated, reductions in capital and operating costs for subsequent plants of a similar design can be expected for the following two reasons:

1. Operating experience and performance data available from the first unit should permit design simplifications, possible increases in core performance, and the use of more rewarding fueling schemes.

2. The market for the reactor components should increase, resulting in the development of improved fabrication techniques for such items as fuel elements and Zircaloy tubing.

A summary of possible cost reductions as a result of these developments is presented in Table 6.1. The figures apply to a 300 MW_e, boiling-D_2O, direct cycle, oxide-fueled reactor. Each of the items can be considered as being independently applicable. If all items were to become effective, a net reduction in power cost of about 1.3 mills/kwh could be expected.

6.2 COST REDUCTION POTENTIAL RESULTING FROM THE DEVELOPMENT PROGRAM

The heavy water-moderated, natural uranium-fueled reactor plant for which the "current status" cost estimates are presented is based on design criteria which have been demonstrated in other reactors or are sufficiently conservative to entail no undue risk. This "current status" reactor could be designed and built starting July 1, 1960. The power generation costs for this reactor in a 300 MW_e plant are given in Table 4.1.

The improvements which are the aim of the 10-year development program are discussed below. It should be noted that these improvements, and the savings resultant from them, are additive to the improvements and cost reductions expected in succeeding generation reactors (Section 6.1). The main items in the future potential development program are summarized in Table 6.2. It is not known whether all of these improvements could be attained by 1970 but the most probable ones are fog cooling, bonded pressure tube joints, and reduced fuel fabrication costs; these three items should reduce the power cost to 6.4 mills/kwh. The other items shown on Table 6.2 (superheat and improved fuel cladding and structural materials) would result in substantial power cost reductions but may not be realized in the prescribed 10-year period. No credit has been taken for these latter items in the future potential economic estimates.

Other improvements which may have major effects on power cost, but for which data are not now available, are plutonium recycle and thorium-U^{233} fuel cycle.

H_2O Fog Coolant

It would be desirable to eliminate the use of D_2O in the cooling circuit of this type of reactor. This would markedly reduce the D_2O inventory charges, eliminate the largest possibility of D_2O leakage and accidental loss, and eliminate any need for specially designed turbine and condensor equipment. An excellent coolant for this application appears to be H_2O "fog" consisting of fine water droplets dispersed in steam. This coolant has been shown to have very high heat removal capabilities and has a low enough density so that neutron absorption by the light water is reasonable even in a natural uranium reactor.

NDA, and CISE and Ansaldo of Italy, under joint USAEC-Euratom sponsorship, are currently conducting a research and development program, including heat transfer and fluid flow experiments, on this coolant for H_2O-moderated reactors. The Canadians have recently initiated a study of fog cooling for D_2O-moderated reactors.

The power cost reductions obtained with fog cooling arise, primarily, from the following effects:

1. reduced D_2O inventory,
2. reduced turbine plant capital investment by use of conventional steam plant equipment without special D_2O seals.

Combined, these factors result in a power cost saving of ≈0.60 mill/kwh.

Nuclear Superheat

The conventional power generating equipment used in fossil-fueled power plants is supplied with superheated steam. Nuclear power plants are attempting to gain the high plant efficiency possible with superheat but have been deterred from doing so by the lack of suitable high temperature fuel materials. In the natural uranium-fueled reactors, the problem is compounded by the necessity of a low neutron cross section, high temperature material. This being the case, it cannot be assumed that nuclear superheat will result from the heavy water reactor 10-year program. As an illustration of the incentive for proceeding with the development of superheat, however, it may be stated that turbine throttle conditions of 1000 psia and 950°F, made possible by beryllium structural materials and cladding, would reduce power cost by 1.7 mills/kwh.

Research and development for FWCNG should contribute measurably to this area. Design steam conditions at the turbine for the prototype plant are: 1450 psia, 950°F.

Improved Zircaloy Materials

Low cross section, good high temperature mechanical properties, and water corrosion resistance are a few of the properties sought in improved clad and structural materials. For example, if improvements can be made in zirconium alloys such that the amount of structural and cladding material in the reactor is cut in half, the fuel burnup could be increased by 25% and power cost would be reduced by 0.32 mill/kwh.

Bonded Zircaloy-to-Stainless Steel Joint

The current development program includes a bonded pressure tube joint which is showing promise of success. With this type of joint, it will be possible to place the joints at the edges of the reactor core, thus shortening the Zircaloy section by about 1/2. This results in a capital cost reduction of $1,700,000, or 0.11 mill/kwh.

Fuel Fabrication Cost Reduction

Each reactor project includes a fuel element development program and the AEC is sponsoring other research projects on fuels. In addition, fuel element manufacturers are constantly improving production techniques. This work will undoubtedly result in substantial fabrication cost reductions. If the current UO_2 fabrication cost is cut from $48.80/kg-U to $40/kg-U, power cost would be reduced by 0.15 mill/kwh.

Table 6 1 — Potential Power Cost Reductions for Succeeding Generation 300 MW_e, Boiling D_2O, Pressure Tube, Cold Moderator, Natural UO_2-Fueled Reactors

(Cost Reductions Refer to Current Reactor Concept, Table 4 1)

Technical Improvement	Effects	Capital Investment	D_2O Investment	Fuel Costs	D_2O Makeup	Operation and Maintenance	Working Capital	Insurance	Total
		Power Cost Reductions, mills/kwh							
1 Improved plant design (based on but not including cost reduction of vapor suppression concept)	a Consolidated equipment arrangement for better utilization of space b Saving in equipment cost and D_2O inventory c Improved fabrication techniques for core and D_2O headering d Estimated cost reduction of 5% in capital and D_2O investment	0 231	0 064	—	0 01	—	—	—	0 305
2 Improved fuel scheduling by on-power refueling	a Fuel burnup increased from 8500 to 10,000 MW-d/metric ton U b Number of control rods reduced by 50% c Cost of refueling machine increases d Design in accordance with concept discussed in NDA 2109 4, Section 3 2 2	(+0 029)	—	0 261	—	—	0 001	—	0 233
3 Zircaloy pressure tube fabrication cost reduced	a Reduction in cost of Zircaloy tubing from \$30 to \$15 per lb as a result of fabricating experience and quantity procurement	0 192	—	—	—	—	—	—	0 192
4 Decreased cost of fuel fabrication	a Reduction in cost of fuel fabrication from \$64 to \$48 80 per kg U as a result of fabricating experience and quantity procurement	0 003	—	0 233			0 045		0 281
5 Vapor suppression in place of containment	a Small steel housing around reactor complex in lieu of containment building as described in Reference S&L-1815	0 096	—	—	—	—	—	—	0 096
6 Increased fuel specific power (cost reductions based on and additive to lower values resulting from above in improvements)	a Fuel element center line temperature limit increased from 4000 to 4500 F b Based on the assumption that fuel burnup, reactivity, and steam conditions remain unchanged, fuel element specific power increases and number of elements decrease	0 083	0 116	(+0 015)	0 019	—	0 013	—	0 216
Total power cost reduction		0 576	0 180	0 479	0 029	—	0 059	—	1 323

Table 6.2 — Potential Power Cost Reductions for 300 MW_e Reactors as a Result of the 10-Year D_2O-Moderated Reactor Development Program

(Cost Reductions Refer to Revised Succeeding Generation Reactor Concept, Table 6.1)

				Power Cost Reductions, mills/kwh								
Technical Improvements	Advantages	Disadvantages and Uncertainties	Basis of Estimate	Capital Investment	D_2O Investment	Fuel Costs	D_2O Makeup	Operation and Maintenance	Working Capital	Insurance	Total	Revised Power Cost
1. H_2O fog-cooled concept (dispersion of liquid drops in steam)	a. Low density of coolant permits substitution of H_2O for D_2O coolant to obtain the following: 1. Reduction in turbine plant capital investment. 2. Reduction in D_2O inventory investment and D_2O makeup.	a. Large steam pumping equipment and special spray equipment required. b. Power coefficient of reactivity may be unfavorable.	a. Lattice spacing increased from 11.1 to 12.8 in. to hold burnup at 10,000 MW-d/metric ton-U. b. Steam conditions remain the same as current boiling D_2O reactor. c. 20% inlet quality, 50% exit quality. d. Cost of steam pumping equipment assumed to offset savings in steam drums and recirculating piping.	0.061	0.457	0.003	0.073	—	—	—	0.594	6.689
2. Improved Zircaloy to steel joints	a. Metallurgically bonded instead of mechanically bonded joints permit shorter Zircaloy pressure tube sections from 48 to 21 ft.		a. Cost of Zircaloy assumed to be $15/lb.	0.108	—	—	—	—	—	—	0.108	7.175
3. Decreased cost of fuel fabrications as result of fuel fabrication development program	a. Lower fuel and working capital costs. b. Fuel costs reduced from $48.80 to $40/kg U including shipping of nonirradiated fuel and losses during fabrication.			—	—	0.129	—	—	0.022	—	0.151	7.132
4. Superheat concept	a. All advantages of fog-cooled concept above. b. Higher cycle efficiency from 28.6 to 34.5% with reheat turbine.	a. Requires beryllium alloy or beryllium alloy insulated pressure tubes. b. Requires beryllium alloy fuel element cladding. c. Burnout and heat transfer data must be developed in order to design the core. d. Control may be more difficult.	a. Single region core-steam enters saturated and leaves at 1500 psi, 950°F. b. Beryllium alloy pressure tubes. c. Beryllium alloy clad fuel elements. d. Burnup assumed to be 15,000 MW-d/metric ton-U. e. Beryllium cost estimated at $150/lb or 2.7 times more costly than Zr.	0.461	0.565	0.491	0.091	0.099	(0.021)	0.022	1.750	5.533
5. Improved Zircaloy structural materials	a. Lower neutron capture cross section loading to increased reactivity and increased burnup to 12,500.	a. Increased fabrication cost.	a. Steam conditions assumed to be the same as reference design. b. All savings assumed to be increased burnup. c. New material assumed to have 1/2 the poisoning effect of reference design Zr-2 pressure tube.		—	0.324	—	—	—	—	0.324	6.959

7. APPENDIX

The sections of this appendix summarize the main parts of the U.S. Heavy Water Reactor Program, the Canadian reactor projects, and the PRTR. The capital cost breakdown for the reactor used in the economic analysis (300 MW_e) is given in Section 7.1.

Characteristics of the reactors in the program, and of several design study reactors, are presented in Table 7.1.

7.1 DESIGN STUDIES OF D_2O-MODERATED POWER REACTORS

E.I. du Pont de Nemours and Company, Inc.

Sargent & Lundy, Engineers

Nuclear Development Corporation of America

Design studies of heavy water-moderated power reactors started at the du Pont Company in 1956. S&L and NDA were brought into the program on an accelerated schedule independently of du Pont in the fall of 1958. From mid-1959 until the present time the three companies have been conducting development work in a cooperative program under prime contracts from the AEC. This work is closely coordinated by the Savannah River Operations Office. Since the three organizations are concerned with a single project, the work will be discussed as a unit.

Objectives

The power reactor design studies had as their objectives (1) the selection of the heavy water-moderated, natural uranium-fueled reactor concept which had the most promise of providing economic power in the near future, and (2) the definition of the development program which should be followed to properly exploit the concept potential.

Concept Description

The most recent studies concerning D_2O-moderated power reactors are reported in SL-1776 and SL-1815. The first of these is a comparison of the direct and indirect cycle plants for a 200 MW_e station. The results of this comparison are shown in Table 7.2 and indicate that the direct cycle has a small economic advantage over the indirect cycle. The direct cycle plant has been selected as the concept that has the greatest development potential for further cost reductions.

SL-1815 presents a 300 MW_e, boiling D_2O-cooled, pressure tube, direct cycle plant that would produce power at a lower unit cost than any other D_2O-moderated reactor plant previously considered. This plant is based on technology currently available and is operable on natural UO_2.

The reactor concept, as presented in SL-1815, has been adapted to this study so as to conform to the site location specified in the ground rules. All economic data presented in this report are

Table 7.1 — Plant Characteristics

	NDA-S&L 200 MW$_e$ Direct Cycle	NDA-S&L 200 MW$_e$ Indirect Cycle	NDA-S&L 70 MW$_e$ Prototype	CVTR	FWCNG Prototype	FWCNG Full Scale	CANDU	NPD-2	PRTR	HWCTR
Category	Boiling D_2O-cooled, pressure tube, direct cycle, cold moderator, natural UO_2-fueled, reactor power plant	Boiling D_2O-cooled, pressure tube, indirect cycle, cold moderator, natural UO_2-fueled, reactor power plant	Boiling D_2O-cooled, pressure tube, direct cycle, cold moderator, natural UO_2-fueled, reactor power plant prototype	Liquid D_2O-cooled, pressure tube, indirect cycle, cold moderator, enriched UO_2-fueled, reactor power plant prototype	Gas-cooled, pressure tube, indirect cycle, cold moderator, enriched UO_2-fueled, reactor power plant prototype	Gas-cooled, pressure tube indirect cycle, cold moderator, enriched UO_2-fueled, reactor power plant	Liquid D_2O-cooled, pressure tube, indirect cycle, cold moderator, natural UO_2-fueled, reactor power plant	Liquid D_2O-cooled, pressure tube, indirect cycle, cold moderator, natural UO_2-fueled, reactor power plant demonstration	Liquid D_2O-cooled, pressure tube, indirect cycle, heat dump, cold moderator, fueled with natural UO_2 and Pu elements, test reactor	Liquid D_2O-cooled, pressure vessel, indirect cycle, heat dump, hot moderator, fueled with natural uranium test region and enriched drive section, test reactor
Heat Balance										
Total reactor power, MW$_t$	790	790	255	62.9	151.6	896	794	89.1	70	50
Gross turbine power, MW$_e$	240	237 8	73 6	19	57.3	320	208.4	22.0	—	—
Net plant power, MW$_e$	224.4	223 6	69.1	17	50	300	200	19.3	—	—
Net plant efficiency, %	28.4	28 3	27.0	28 (with oil-fired superheater)	33.2	33.5	25.2	21.7	—	—
Turbine Cycle Conditions										
Throttle temp, °F	510	480	510	725	950	950	419	448	—	—
Throttle pressure, psia	765	566	765	415	1450	1465	305	415	—	—
Total steam flow, lb/hr	3.27×10^6	2.88×10^6	1.059×10^6	200,000	349,265	2×10^6	2.75×10^6	0.3×10^6	—	—
Condenser back pressure, in. Hg A	1.5	1.5	1.5	1.5	2.5	1.5	1.5	1.5	—	—
Final feedwater temp, °F	387	364	387	280.9	317	320.2	305	300	—	—
No. of feedwater heating stages	4	5	4	3	3	3	4	4	—	—
Reheat temp, °F	—	—	—	—	950	950	—	—	—	—
Reheat pressure, psia	—	—	—	—	215	215	—	—	—	—
Reheat steam flow, lb/hr	—	—	—	—	330,255	1,974,500	—	—	—	—
Reactor Description										
Reactor vessel										
Type	Calandria	Calandria	Calandria	Cylindrical tank	Calandria			Calandria	Calandria	Pressure vessel
ID, ft	18.4	18.4	16	9.5	13	20 ft w × 30 ft l	20.16	14.66 at center, 12 at ends	11	7
Inside height, ft	20.2	20.2	15.6	14.75	24 (includes 5 ft H_2O shield tanks at each end)	18	18.6	12.58 long	9.58	29 ft
Wall thickness, in. (cylindrical portion)	0.375	0.375	0.375	0.25	0.75	0.75			0.25	3.25 min
Material	Al	Al	Al	SS	SS	SS	Al	Al	Al	1/4 in. SS-clad carbon steel
Design pressure, psig	15	15	15	12	80		Atm	Atm	7.5	1500
Design temp, °F	215	215	150	215	~300			~150	205	600
Reactor core geometry										
Active diam, ft	16.4	16 4	12	6.93	10.83	21.6	18.1	11	7	4
Active height, ft	17.7	17.7	11 1	8.0	12	18	16.6	12.6	7.33	10
Lattice arrangement	Triangular	Triangular	Triangular	Rectangular	Rhomboid	Square	Square	Square	Triangular	Central test section triangular surrounded by enriched driver section
Lattice spacing, in.	11.1	11.1	11.1	8 0 × 6.5	8 × 10 3	10	12.2	10.25	8	7 (test region)
Total no. of lattice positions	287	287	152	42 U tubes	210	528	252	132	84	
Total no. of fueled positions	268	268	133	84	192	528	252	132	UO_2 – 42-75, Pu-Al – 10-43	Test – 12, Driver – 24
Reflector										
Material	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	D_2O	H_2O, D_2O	D_2O	D_2O
Axial thickness, ft	1	1	2	1.67	1	1	1			—
Radial thickness, ft	1	1	2	1.0	1	1	1	1.08 – H_2O, 1.8 – D_2O	1.95	—
Fuel elements										
Geometry	Rods	Rods	Rods	Rods	Rods	Rods	Rods	Rods	Clustered rods (Mark I), concentric cylinders (Mark II)	Tube (driver), tube or rods (test)
Fuel material	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	UO_2	Mark I – UO_2 or PuAl, Mark II – UO_2	9.3% oralloy in Zr (driver fuel), natural U (test fuel)
Fuel meat thickness, in.	0.500	0 500	0.500	0.43	0.5 OD w/0.2, BeO core	0.5	0.58	0.937	Mark I – 0.504 diam, Mark II – 0.548 diam centered – 0.350 in. center tube – 0.310 in. outer tube	0.137 driver fuel, 0.267 test fuel
Clad material	Zr-2	Zr-2	Zr-2	Zr-2	BeO	Beryllium	Zr-2	Zr-2	Zr-2	Zr-2, Zr-4
Clad thickness, in.	0.025	0 025	0.025	0.023	0 030	0.060	0 02	0 025	0.030 (Mark I), 0.060 (Mark II)	0.015
Clad gap, in.	0.005	0 005	0.005	0 003	0 003	0.000				
Gap filler material	He	He	He	He						
Fuel element assembly										
Total no. of assemblies	536 (2 per lattice position)	536 (2 per lattice position)	266 (2 per lattice position)	84	768	—	4500	1056 (8 per lattice position)	85	36
No. of elements (rods) per assembly	37	37	37	19	19	19	31	7	19 rods (Mark I), 1 rod, 2 concentric tubes (Mark II)	Driver fuel – 1 tube with stainless steel cross with 0.6% boron test fuel – 1 tube
Active length, ft	8.35	8.35	5.55	8	(11 25 total)	(17.6 total)			7.33	9.4
End fitting materials	Zr-2	Zr-2	Zr-2	Zr-2	Beryllium	Beryllium	Zr-2	Zr-2	Zr-2	
Calandria tubes										
Material	Al	Al	Al	—	—	—	Al	Al	Al	—
ID, in.	5.724	5.724	5.724	—	—	—	5.25	4	4.12	—
Wall thickness, in.	0.060	0.060	0.060	—	—	—	0.0625	0.052	0.065	—
Pressure tubes										
Material	Zr-2	Zr-2	Zr-2	Zr-4	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2	Zr-2 (fuel housing)
ID, in.	4.650	4.650	4.650	3.53	5.01	5.260	4.50	3.25	3.250	2.9
Wall thickness, in	0 162	0.162	0.162	0.253	0.120	0.120	0 250	0.163	0.154	0.030
Coolant-moderator insulation										
Material	Air	Air	Air	D_2O	CO_2	CO_2	He	Air	He	—
Thickness, in	0 375	0 375	0.375	0.044	0.07	0.07	0 125	0.212	0.28	—
No of insulating gaps	1	1	1	4	2	2	1	1	1	—
Gap separators	Calandria and pressure tube annulus	Calandria and pressure tube annulus	Calandria and pressure tube annulus	Zr-2 baffles	Calandria and pressure tube annulus, flow liner – 0.02 in. Zr-Cu-Mo; radiation liner – 0 02 in Zr-Cu-Mo, gap separator – 0 02 in Zr-2	Calandria and pressure tube annulus, flow liner – 0.02 in. Zr-Cu-Mo, radiation liner – 0.02 in. Zr-Cu-Mo, gap separator – 0.02 in. Zr-2	Calandria and pressure tube annulus	Calandria and pressure tube annulus	Calandria and pressure tube annulus	—
Separator thickness, in.	—	—	—	0.021			—	—	—	—

Note Blank spaces indicate data not available.

Table 7.1 — (Continued)

	NDA-S&L 200 MW_e Direct Cycle	NDA-S&L 200 MW_e Indirect Cycle	NDA-S&L 70 MW_e Prototype	CVTR	FWCNG Prototype	FWCNG Full Scale	CANDU	NPD-2	PRTR	HWCTR
Reactor Description (Contd.)										
Control										
Method	Rods	Rods	Rods	Rods	Vertical rods in pressure tube positions		D_2O moderator level and on-power refueling	D_2O moderator level and on-power refueling	Moderator level and shim rods	Rods and burnable poison driver fuel
Absorber material				Boron SS			—	—	Inconel	Black – SS with boron, gray – SS
No. of control elements	19	19	19	32	18		—	—	54	12 black; 3 gray
Type of drive	Motorized	Motorized	Motorized	Motorized	Motorized			—	Motorized	Motorized
Size				10 in. perimeter				—		
Material inventories										
Total fuel loading, *metric tons of uranium*	59.7	59.7	17.3	4.04	15.4	69.0	54.6	15	Mark I – 12 lb UO_2/element – 70 lb Pu-Al element, Mark II – 141 lb/UO_2 element	
% enrichment	0.72	0.72	0.72	1.6	1.2	0.99	0 72	0.72		
D_2O (core and reflector), tons	140	140	95.7		42.4	207				
Reactor Performance										
Primary coolant										
Coolant material	D_2O	D_2O	D_2O	D_2O	CO_2	CO_2	D_2O	D_2O	D_2O	D_2O
Coolant outlet temp, °F	515	515	515	555	1050	1050	525	530	530	554
Coolant inlet temp, °F	498	498	498	505	550	550	430	485	478	
Coolant pressure, psig				1500	500	540	1000	1000	1025	1500
Coolant flow, lb/hr	23.4×10^6	23.4×10^6	7.54×10^6	3.3×10^6	2.41×10^6	19.66×10^6	25×10^6	5.14×10^6	4.59×10^6	4.8×10^6
Avg core velocity, ft/sec	15.1 at inlet	15.1 at inlet	8 at inlet	22	70			15.7	11.1	14
Heat transfer										
Max fuel center temp, °F	4000	4000	4000	4230	3100	4500	4000	4000		
Max fuel clad temp, °F	540	540	540	587	1450		540	547		
Max core heat flux, Btu/hr-ft²	2.94×10^5	2.94×10^5	2.4×10^5	3.49×10^5	1.04×10^5		3.3×10^5	1.58×10^5	3.3×10^5	
Burnout heat flux, Btu/hr-ft²				1.40×10^6						
Power to coolant, MW_t	733	733	237	56	138	827.4		83.3	66.4	50 MW
Power to moderator and reflector, MW_t	57	57	18	6.6	13.6	78.6		6.6	2.8	
Avg moderator temp, °F	155	155	120	155	232.5	231		150	143	
Secondary coolant										
Coolant material	—	H_2O	—	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O	H_2O
Method of heat transfer	—	Steam generator	—	Steam generator	Steam generator	Steam generator	Steam generator	Steam generator	Steam generator	Steam generator
Coolant outlet temp, °F	—	480	—	487	950	950	419	448	450	470
Coolant inlet temp, °F	—	475	—	282	317	320.2	305	415	228	104
Coolant pressure, psig	—	551	—	605	1450	1450	290	400	425	545
Coolant flow, lb/hr	—	2.88×10^6	—	201,000	349, 265	2×10^6	2.75×10^6	0.3×10^6	214,050	3.16×10^6
Nuclear										
Radial max to avg flux	1.87	1.87	1.56	1.25	1 3					
Axial max to avg flux	1.45	1.45	1.25	1.38	1.57				1.39	
Bundle max to avg flux	1.1	1.1	1.1	1.25	1.10				1.20	
Localized power variations				1.25						
Max to avg thermal neutron flux	2.88	2.88	2.14	2.7			1.32 at center	2.25		
Fuel cycle										
Management	Off-power, 4-zone radial shift axial repositioning	Off-power, 4-zone radial shift axial repositioning	Off-power, 5-zone radial shift axial repositioning	Off-power batch or 3-zone radial shift	On-power continuous	On-power continuous	On-power continuous	On-power continuous	Off-power batch	Off-power batch
Avg burnup, MW-d/metric ton-U	7500	7500	6100	Batch – 12,400, Zone – 23,700	10,000	15,000	8100	5400	5-8000	3000 (test), 50% (driver)
Containment										
Design criteria	Vapor containment	Vapor containment	Vapor containment	Vapor containment						
Type	Steel shell	Steel sheel	Steel shell	Reinf. concrete, steel lined	Vapor containment	Vapor containment			Vapor containment	Vapor containment
Geometry	Cylindrical	Cylindrical	Cylindrical	Cylindrical	Steel shell	Steel sheel		Reactor vault	Steel shell	Steel shell
Dimensions	135 ft × 185 ft high	135 ft × 185 ft high	108 ft × 166 ft high	58 ft × 119 ft high	87 ft × 134 ft high	160 ft		223 × 55 × 38	80 × 121.5 ft high	70 ft × 125 ft high
Design pressure, psig	28	15	10.8	20	27	15		10	15	24
Material	Steel	Steel	Steel	3/16 in. steel plate, 24 in. concrete	Carbon steel	Steel		Reinforced concrete	Steel	Steel and reinforced concrete
References	43	43	14	56	SROO	52, 53	57	12	54	55

Note Blank spaces indicate data not available

Table 7.2 — Direct and Indirect Cycle Data

Item	Direct	Indirect
Reactor gross thermal output	790	790
Plant net electrical output	224.4	223 6
Plant efficiency, net	28.4	28.3
Turbine throttle temperature, °F	510	480
Turbine throttle pressure, psia	765	566
Capital investment, $	57,141,000	62,050,000
D_2O inventory, $	16,500,000	12,970,000
Capital charges, mills/kwh	5.10	5.55
D_2O inventory charges	1.31	1.03
Fuel costs	2.12	2.13
Operating, maintenance, and insurance	1.26	1.19
Total power cost	9.79	9.90

based on this reactor concept. The 100 MW_e plant is a scaled version of the 300 MW_e plant. Plant characteristics for both plants may be found in Table 7.3; capital costs are shown according to the AEC system of accounts in Table 7.4; nuclear fuel cycle costs are given in Table 7.5. A brief description of the 300 MW_e plant is given below. More detailed information on the complete plant will be found in SL-1815.

Reactor

The 300 MW_e boiling D_2O, pressure tube reactor consists of an aluminum calandria, containing cold D_2O moderator, which is pierced by 394 calandria tubes (369 fueled lattice positions and 25 control rods) arranged in an equilateral triangular lattice array with an 11.1 in. spacing. Subcooled D_2O liquid enters at the bottom of the pressure tubes and leaves the top as a 14% quality steam-water mixture. The steam-water mixture from the pressure tubes flows to two steam drums where the steam is separated for use directly in the turbine. Feedwater from the turbine plant is returned to the steam drums at 387°F where it subcools the separated D_2O liquid and the resulting mixture returns to the lower header at 499°F for recirculation.

The aluminum calandria is 20.6 ft in diameter and 23.1 ft in height. During normal operation, D_2O fills the calandria shell, except for a helium cover gas space under the upper tube sheet, and is circulated and cooled to maintain an average moderator temperature of 155°F.

The Zircaloy-2 pressure tubes, which contain the coolant, have an OD of 4.97 in., a wall thickness of 0.16 in., and are approximately 45 ft long. The pressure tubes pass through the calandria tubes and are connected, by mechanical joints, to stainless steel coolant flow distribution piping above and below the core. The coolant enters the reactor through a header and pigtail distribution system, located in the lower header room, which is designed to accommodate thermal expansion of the pressure tubes. In the upper header room, the coolant passes into a cross core headering system which supports the pressure tubes. It is then carried to two steam drums.

Table 7.3 — 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 7.3 — (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

Table 7.4 — Capital Cost Estimate – Boiling D_2O, Pressure Tube, Direct Cycle Power Reactor Plants

ACCT. NO.	DESCRIPTION	110 Gross MW_e TOTAL	325 Gross MW_e TOTAL
	Summary of Direct Costs		
20	Land and Land Rights	360,000	360,000
21	Structures and Improvements	4,144,650	6,585,760
22	Reactor Plant Equipment	12,528,550	22,717,630
23	Turbine Generator Units	6,522,450	15,959,190
24	Accessory Electric Equipment	1,512,000	2,046,090
24	Miscellaneous Power Plant Equipment	384,400	503,750
	Total Direct Construction Costs	$25,422,050	$48,172,420
	Indirect Construction Costs		
	General and Administrative		
	At 13.5 for 110 MW_e and 12.5 for 325 MW_e	3,431,977	6,021,553
	Sub-Total	$28,854,027	$54,193,973
	Engineering, Design and Inspection		
	At 15 for 110 MW_e and 14.6 for 325 MW_e	4,328,104	7,912,320
		$33,182,131	$62,106,293
	Start-Up Costs		
	At 4½ months of Operating Costs	602,000	941,250
	Sub-Total	$33,784,131	$63,047,543
	Contingencies at 10	3,378,413	6,304,754
	Sub-Total	$37,162,544	$69,352,297
	Interest During Construction		
	At 6.7 for 110 MW_e and 8.1 for 325 MW_e	2,489,890	5,617,536
	Total Capital Cost, Dollars	$39,652,434	$74,969,833
	Net Power, kw_e	103,000	318,900
	Unit Capital Cost, Dollars per Net kw_e	385	235

Table 7.4 — (Continued)

ACCT. NO.	DESCRIPTION	110 Gross MW_e TOTAL	325 Gross MW_e TOTAL
20	LAND AND LAND RIGHTS		
201	LAND AND PRIVILEGE ACQUISITION		
	.1 Land Owned in Fee	360,000	360,000
	TOTAL ACCT. 20	$ 360,000	$ 360,000
21	STRUCTURES AND IMPROVEMENTS		
211.1	ACCESS RODS FOR PERMANENT USE	0	0
211.2	GENERAL YARD IMPROVEMENTS		
	.21 Grading and Landscaping		
	.211 Clearing Site	4,000	4,000
	.212 Fill and Grading (On Site)	32,000	32,000
	.213 Landscaping	2,500	4,000
	.22 Roads, Sidewalks, and Parking		
	.221 Roadways	13,000	13,000
	.222 Sidewalks	2,500	2,500
	.223 Parking Area	9,000	13,000
	.23 Fence	12,500	16,000
	.24 Outside Water – On Site	6,000	6,000
	.25 Sewers and Drainage System		
	.251 Sewer System	35,000	43,000
	.252 Yard Drainage	17,000	17,000
	.26 Roadway and General Yard Lighting		
	.27 Test Borings	9,500	9,500
	.28 Miscellaneous Other Items	8,500	10,000
211.3	RAILROAD TRACK ON AND OFF SITE		
	.31 Railroad Track on Site	28,500	34,000
	.32 Railroad Track Off Site	300,000	300,000
	TOTAL ACCT. 211	$ 480,000	$ 504,000
212	BUILDINGS		
212-A	TURBINE ROOM BUILDING INCL. CONTROL ROOM, OFFICES, MACHINE SHOP, ETC.		
	.1 Excavation, Backfill, and Disposal – Incl. Sheeting, Dewatering, Etc. As Required	44,400	73,000
	.3 Substructure Concrete – Incl. Concrete, Reinforcing Steel, Form Work, Waterproofing, Misc. Anchor Bolts, Sleeves, Etc., Embedded in Concrete	112,000	184,500
	.4 Superstructure		
	.41 Concrete Reinforcing, Forms		
	.421 Structural and Girt Steel and Stack		
	.422 Misc. Steel, Stairs, Railing, Grating, Etc.		
	.43 Insulated Metal Panel Siding		
	.44 Roof Slabs, Insulation, Roofing, Flashing		

Table 7.4 — (Continued)

ACCT. NO.	DESCRIPTION	110 Gross MW_e TOTAL	325 Gross MW_e TOTAL
212-A	.45 Roof Slabs, Insulation, Roofing, Flashing	523,000	805,500
	.46 Windows, Doors, Hardware, Etc.		
	.47 Glazed Tile Plaster, Suspended Ceiling		
	.48 Floor Finishes		
	.49 Painting		
	.65 Elevator Enclosure		
	.5 Stacks		
	.6 Service Work		
	.61 Plumbing and Drainage		
	.611 Plumbing		
	.612 Drainage	30,000	32,000
	.613 Domestic Water Tank Hot	1,200	1,200
	.614 Domestic Water Tank	3,000	3,800
	.615 Duplex Sump Pump Incl. Motors	1,500	2,500
	.616 Pumps and Motors for D_2O	1,900	2,500
	.62 Heating System		
	.621 Heating Boiler and Assoc. Equip. Incl. Fuel Oil Storage Tank, Transfer Pump and Motor		
	.63 Ventilating System	59,200	103,120
	.64 Air Conditioning		
	.65 Elevator	15,000	15,000
	.651 Elevator Enclosure	Incl. 212A-.4	
	.66 Lighting and Service Wiring	35,600	49,000
	.7 Miscellaneous		
	.71 Miscellaneous Equipment Foundations	31,500	32,000
	TOTAL ACCT. 212-A	$ 858,800	$ 1,303,620
212-B	WASTE DISPOSAL BUILDING AND ASSOCIATE STRUCTURES		
	.1 Excavation, Backfill, and Disposal – Incl. Sheeting, Dewatering, Etc. As Required		
	.3 Substructure Concrete – Incl. Concrete, Reinf., Forms, Waterproofing, Misc. Bolts, Sleeves, Etc. Embedded in Concrete		
	.4 Superstructure		
	.41 Concrete, Reinforcing, Forms, Etc.		
	.421 Structural and Girt Steel	182,000	208,000
	.422 Miscellaneous Steel, Stairs, Railing, Grating, Etc.		
	.43 Insulated Metal Siding		
	.44 Roof Slabs, Insulation, Roofing, Flashing		
	.45 Interior Partitions		
	.46 Windows, Doors, Hardware, Etc.		
	.48 Floor Finishes		
	.49 Painting		
	.61 Plumbing and Drainage		

Table 7.4 — (Continued)

ACCT. NO.		DESCRIPTION	110 Gross MW_e TOTAL	325 Gross MW_e TOTAL
212-B	.5	Stacks		
		.51 Concrete Chimney Foundation	28,000	28,000
		.52 Concrete Chimney	6,000	6,000
	.6	Service Work		
		.61 Plumbing and Drainage	Incl. 212-B-.4	
		.611 Duplex Sump Pump	1,250	1,340
		.65 Lighting and Service Wiring	12,000	13,400
	.7	Miscellaneous		
		.71 Miscellaneous Equipment Foundations	8,500	8,500
		.72 Miscellaneous Other Items	6,000	6,000
		TOTAL ACCT. 212-B	$ 243,750	$ 271,240
212-C		FUEL HANDLING BUILDING AND TRANSFER TUNNEL		
	.1	Excavation, Backfill, Etc.		
	.3	Substructure Concrete – Incl. Concrete, Reinforcing, Forms, Misc. Bolts, Sleeves, Etc., Embedded in Concrete		
	.4	Superstructure		
		.41 Concrete, Reinforcing, Forms, Etc.		
		.421 Structural and Girt Steel	250,000	364,500
		.422 Misc. Steel, Ladder, Railings, Etc.		
		.43 Insulated Metal Siding		
		.44 Roof Slabs, Insulation, Roofing, Flashing		
		.45 Interior Partitions		
		.46 Windows, Doors, Hardware, Etc.		
		.48 Floor Finishes		
		.49 Painting		
	.6	Service Work		
		.61 Plumbing and Drainage		
		.62 Heating System		
		.63 Ventilating System		
		.64 Air Conditioning		
		.66 Lighting and Service Wiring	12,000	15,400
		TOTAL ACCT. 212-C	$ 262,000	$ 379,900
212-D		D_2O DISTILLATION STRUCTURES		
	.1	Excavation, Backfill, Etc.	13,500	13,500
	.3	Substructure		
	.4	Superstructure		
	.6	Service Work		
	.7	Miscellaneous		
		.71 Miscellaneous Other Items	1,500	1,500
		TOTAL ACCT. 212-D	$ 15,000	$ 15,000

Table 7.4 — (Continued)

ACCT. NO.			DESCRIPTION	110 Gross MW_e TOTAL	325 Gross MW_e TOTAL
212	BUILDINGS				
212-E	MISCELLANEOUS BUILDINGS				
	a.		Gate House	9,000	9,000
	b.		Warehouses	94,000	121,000
	c.		Oil Pump House	400	400
	d.		Field Construction Office and Other Construction Buildings, Incl. Parking Area	37,000	53,000
	e.		Foundation and Berm for 50,000 Gal. Oil Tank	1,500	1,500
	f.		Foundation – Day Tank for Heating Boiler	400	500
	g.		Miscellaneous Other Items	4,500	5,700
			TOTAL ACCT. 212-E	$ 146,800	$ 191,100
			TOTAL ACCT. 212	$ 1,526,350	$ 2,160,860
219	REACTOR CONTAINER STRUCTURE				
	.1		Excavation, Backfill, and Disposal – Incl. Sheeting, Dewatering, Etc. As Required	40,500	120,000
	.3		Substructure Concrete Outside Containment Vessel – Incl. Concrete, Reinforcing, Forms, Waterproofing, Miscellaneous Anchor Bolts, Sleeves, Etc. Embedded in Concrete	102,000	161,000
	.4		Superstructure		
		.41	Structural Steel Supports – Inside Containment Shell	39,000	144,000
		.42	Containment Shell		
			a. Containment Shell Steel		
			b. Shell Supports		
			c. Air Lock Doors	626,000	1,220,000
			d. Water Tank at Top of Shell		
			e. Testing		
		.43	Insulation of Shell		
			a. Insulation of Shell Above Grade	115,000	144,000
			b. Coating Below Grade	2,700	6,000
		.45	Shielding Doors for Compartments Around Reactor	94,000	107,500
		.46	Concrete – Inside Containment Vessel	690,000	1,330,000
		.461	Flexcell Liners	15,800	23,500
		.48	Miscellaneous Steel		
			a. Gallery	27,000	34,500
			b. Misc. Steel Anchored to Concrete	6,000	19,000
			c. Spent Fuel Transfer Tube (Stainless Steel)	7,500	10,000
		.481	Liner Plates		
			a. Upper Header Room (Stainless Steel)	51,000	123,000
			b. Moderator D_2O Storage Tank (Stainless Clad)	9,700	23,400

Table 7.4 — (Continued)

ACCT. NO.			DESCRIPTION	110 Gross MW_e TOTAL	325 Gross MW_e TOTAL
219			c. Coolant D_2O Storage Tank (Stainless Clad)	8,300	20,100
			d. Reactor, Misc. Drain Tank Pit (Stainless Clad)	3,000	7,000
			e. Duplex Pump Pits (Stainless Clad)	2,000	4,500
			f. Endless Cup Conveyor (Stainless Clad)	17,000	37,500
		.49	Painting, Coating, Insulation		
		.491	Painting	37,000	99,000
		.492	Fresh Fuel Storage Well Coating	3,500	3,500
		.493	Lower Header Room Insulation	4,500	11,000
	.6		Service Work		
		.61	Plumbing and Drainage	11,000	12,000
		.611	Duplex Sump Pump Incl. Drives	1,500	3,000
		.612	Duplex Sump Pumps Inc. Drives for D_2O	1,900	3,000
		.62	Heating System		
		.63	Ventilating System	80,000	100,000
		.64	Air Conditioning		
		.65	Elevators	20,000	20,000
		.651	Elevator Enclosure	10,000	10,000
		.66	Lighting and Service Wiring	38,400	39,400
	.7		Miscellaneous		
		.72	Miscellaneous Other Items	44,000	85,000
			TOTAL ACCT. 219	$ 2,108,300	$ 3,920,900
			TOTAL ACCT. 21	$ 4,114,650	$ 6,585,760
22			REACTOR PLANT EQUIPMENT		
221			REACTOR EQUIPMENT		
	.1		Reactor Vessel		
		.11	Calandria Shell	482,000	713,000
		.12	Lattice Parts, Tubes, Seals, Plugs, Etc.	2,495,000	5,280,000
		.13	Headers and Pigtails	951,000	1,745,000
		.14	Misc. Reactor Vessel Parts	594,000	1,081,000
	.2		Control Rod Mechanism		
		.21	Control Rods	880,000	1,410,000
		.22	Housing and Shrouds		
		.23	Drive Mechanisms		
	.3		Shielding		
		.31	Thermal Shields	276,000	323,000
		.32	Neutron Shields	572,500	876,000
		.33	Lower Gamma Shield	82,000	157,500
		.341	Shield Cooling Pumps and Drives	5,000	5,600
		.342	Shield Cooler	7,000	18,300
		.343	Shield Cooler Tank	1,400	1,450
		.344	Full Flow Filters	2,900	6,100

Table 7.4 — (Continued)

ACCT. NO.		DESCRIPTION	110 Gross MW_e TOTAL	325 Gross MW_e TOTAL
221	.4	Cooling Facilities and Equipment		
	.42	Reactor Shut-Down Cooling		
	.421	Shut-Down Cooler	19,000	28,000
	.422	Shut-Down Condensate Pumps	41,000	109,500
	.423	Shut-Down Cooling Pumps Incl. Diesel Drive Unit	6,000	15,200
	.44	Component Cooling		
	.441	Service Water Pumps	70,000	170,000
	.442	Service Water Strainers	12,000	34,000
	.6	Moderator and Reflector		
	.61	Moderator		
	.611	Moderator Circulating Pumps and Motors	29,000	36,000
	.612	Moderator Cooler	50,000	126,000
	.613	Moderator Level and Control Pumps and Motors	2,800	3,800
	.614	Moderator Level Indicating and Fill Standpipe	2,000	3,450
	.615	Moderator Purification Booster Pumps and Motors	2,500	2,950
	.616	Moderator Purification Pre-Filters, Ion Exchangers, Strainers	21,500	56,000
	.63	D_2O Recovery and Air Make-Up System		
	.631	Absorber and Refrigerating Unit	23,000	64,000
	.632	Radioactive Gas Compressor Motor and Aftercooler	8,200	13,500
	.633	High Pressure Gas Storage Tanks	22,000	53,000
	.634	Oxygen Cylinder	150	150
	.635	Vent Gas Heater, D_2O Recombiner, D_2O Vapor Condenser, Refrigerating Unit	6,700	14,800
	.636	Vent Gas Booster Fans and Motors	13,000	13,000
	.64	Moderator Purge Gas System	11,000	13,200
	.65	D_2O Storage and Make-Up Facilities		
	.651	Helium Cylinders, Storage and Manifolds	1,950	5,700
	.652	Accumulator	8,000	22,300
	.653	High Pressure Injection Seal Pumps and Motors	6,400	8,300
	.654	D_2O Moderator Trans. Pumps and Motors	3,500	7,900
	.7	Miscellaneous Items Incl. Painting	23,000	57,000
		TOTAL ACCT. 221	$ 6,731,500	$12,484,700
222		HEAT TRANSFER SYSTEMS		
	.1	Primary Coolant System		
	.11	Pumps		
	.111	Main Pumps	254,000	710,000
	.121	Main Coolant Piping, Valves, and Valve Operating Systems	415,000	886,000
	.122	Insulation Main Coolant Piping	16,000	31,000
	.13	Steam Drums and Separator	497,000	971,000

Table 7.4 — (Continued)

ACCT. NO.			110 Gross MW_e TOTAL	325 Gross MW_e TOTAL
222	.4	Coolant Charging and Discharging System		
		.411 Tanks	Incl. 219, .481	
		.412 D_2O Storage and Make-Up Pumps	6,500	17,700
		.42 Coolant Purification Equipment		
		.421 Demineralizers, Strainers and Filters and Ion Exchangers	70,000	186,000
		.422 Heat Exchangers		
		a. Purification System Regenerative Heat Exchanger	9,200	11,000
		b. Purification System Cooler	13,400	31,600
		c. Distillation System Condensers	10,800	10,800
		d. Distillation System Reboiler Heat Exchanger	1,520	1,520
		e. Distillation Columns	50,000	50,000
		.423 Pumps		
		a. Purification System Booster Pump	32,000	57,500
		b. Spent Resin Transfer Pump and Motor	1,050	1,050
		c. Vacuum Pumps Incl. Motors	4,650	4,650
		d. Reflux Pumps	3,820	3,820
		e. Bottom Pumps	1,560	1,560
		.424 a. Tanks (D_2O)	2,000	2,260
		b. Tanks (H_2O)	500	585
		c. Tanks (Filter Precoat)	800	800
		d. Resin Fill and Disch. Deuterizing and Filter Backwash Column	2,800	2,800
		e. Overhead Tank	1,200	2,800
	.5	Coolant Receiving and Storage Facilities		
		.51 Tanks		
		.511 Misc. D_2O Drain Tank	900	1,700
		.512 Column Blowdown Tank	3,000	3,000
		.52 D_2O Coolant Transfer Pump	3,500	10,300
	.6	Miscellaneous Unlisted Items for Distillation Plant Incl. Painting	7,000	17,000
		TOTAL ACCT. 222	$ 1,480,000	$ 3,016,445
223		FUEL HANDLING AND STORAGE FACILITIES		
	.1	Fuel Handling Crane	45,200	54,300
		.22 Fuel Handling System	623,000	1,056,000
		.23 Television and Special Lighting Equipment	14,500	15,600
		.24 Misc. Tools and Equipment	7,000	10,000
	.3	Spent Fuel Cooling Equipment and Inspection Facilities		
		.31 Fuel Storage Pool Heat Exchanger	4,200	6,500
		.32 Fuel Storage Cooling Pump	900	960
	.4	Shipping Casks and Cars		
		.41 Shim and Safety Rod Cars	79,200	118,000
		TOTAL ACCT. 223	$ 1,122,000	$ 1,817,360

Table 7.4 — (Continued)

ACCT. NO.	DESCRIPTION	110 Gross MW_e TOTAL	325 Gross MW_e TOTAL
225	RADIOACTIVE WASTE TREATMENT AND DISPOSAL FACILITIES		
	.1 Liquid Waste		
	.11 Drain and Storage Tanks		
	.111 New Resin Storage Tank	900	960
	.112 Caustic Tank and Heater	6,500	6,550
	.113 Acid Tank	6,400	6,400
	.114 Cellulose Fill Tank and Mixer	1,000	1,020
	.115 Neutralizer Tank and Mixer	1,500	2,320
	.116 Waste Collector Tank	1,600	4,600
	.117 Waste Hold-Up Tank	10,000	19,000
	.118 Resin Regenerating Tank	1,000	3,300
	.119 Misc. D_2O Drain Tank	1,500	4,550
	.120 Degraded D_2O Storage Tank	50	50
	.12 Ion Exchangers		
	.121 Mixed Bed Exchanger	30,000	30,600
	.13 Filters and Separators		
	.131 Cellulose Filters	150	140
	.14 Pumps		
	.141 Waste Collector Drain Pump	5,300	6,600
	.142 Waste Hold-Up Pumps	2,160	2,160
	.143 Caustic Transfer Pumps	1,900	1,960
	.144 Acid Transfer Pumps	1,900	1,960
	.145 Resin Storage Pumps	2,800	2,875
	.146 Filter Precoat Pumps	1,700	1,740
	.147 Waste Neutralizer Pumps	2,000	2,140
	.2 Gaseous Wastes		
	.21 Waste Gas Blower Incl. Motor Drives	4,900	4,900
	.4 Miscellaneous Other Items Incl. Equipment Painting	7,500	17,900
	TOTAL ACCT. 225	$ 90,700	$ 121,725
226	INSTRUMENTATION AND CONTROL		
	.1 Primary Plant Control		
	.2 Heat Transfer Systems		
	.3 Reactor Safety System	440,000	448,100
	.4 Radioactive Waste System		
	.5 Radioactive Monitoring		
	.6 Steam Generator Controls		
	.7 Control and Instrument Piping and Tubing		
	.8 Electrical Connections, Etc.	120,000	140,900
	TOTAL ACCT. 226	$ 560,000	$ 589,000

Table 7.4 — (Continued)

ACCT. NO.	DESCRIPTION	110 Gross MW_e TOTAL	325 Gross MW_e TOTAL
227	FEEDWATER SUPPLY AND TREATMENT SYSTEMS		
	.1 Raw Water Supply Systems		
	.2 Purification and Treatment		
	.21 Demineralized Water Transfer Pumps	950	2,300
	.22 Demineralizing Equipment	20,000	25,000
	.23 Condensate Filters	34,200	86,000
	.3 Feedwater Tanks		
	.31 Demineralized Water Tank	5,500	9,000
	.4 Feedwater Heaters		
	.41 Feedwater Heaters "A," "B," "C," and "D"	274,000	605,000
	.5 Boiler Feed Pumps Incl. Drives	255,000	739,000
	TOTAL ACCT. 227	$ 589,650	$ 1,466,300
228	STEAMS, CONDENSATE, AND FEEDWATER PIPING INCL. ALL OTHER PIPING FOR REACTOR PLANT CRIBHOUSE, DISTILLATION, WASTE DISPOSAL, AND FUEL HANDLING SYSTEMS		
	.1 Piping	1,665,000	2,842,600
	.2 Insulation-Piping	251,000	364,500
	.3 Painting-Piping	10,500	15,000
	TOTAL ACCT. 228	$ 1,926,500	$ 3,222,100
	TOTAL ACCT. (22)	$12,528,550	$22,717,630
23	TURBINE GENERATOR UNITS		
231	TURBINE GENERATOR		
	.1 Turbine Generator Foundation		
	.11 Foundation	90,000	167,000
	.12 Electrical	7,000	7,110
	.2 Turbine Generator Unit Complete with Accessories and Generator	4,800,000	12,187,000
	.3 Reserve Exciter	100,000	231,100
	.4 Turbine Oil Storage Tank	5,300	6,300
	.5 Turbine Oil Filter Pump	500	680
	.6 Turbine Oil Purifier	3,750	5,250
	Turbine Oil Filter	200	300
	.7 Miscellaneous Unlisted Items Incl. Painting of Equipment		
	TOTAL ACCT. 231	$ 5,043,950	$12,667,040

Table 7.4 — (Continued)

ACCT. NO.	DESCRIPTION	110 Gross MW_e TOTAL	325 Gross MW_e TOTAL
232	CIRCULATING WATER SYSTEM		
	.1 Pumping and Regulating System		
	.11 Vertical Mixed Flow Circulating Water Pumps, Incl. Motor Drives	150,000	432,000
	.12 Traveling Screens, Etc.		
	.121 Traveling Screen Complete Incl. Motor Drive	40,000	118,800
	.122 Screen Wash Pump and Motor Drive	3,200	5,450
	.2 Circulating Water Lines		
	.21 Supply		
	.211 Cribhouse		
	.2111 Substructure		
	.2112 Superstructure	120,000	306,000
	.2113 Structural Steel		
	.212 Seal Well	40,000	45,000
	.213 Circ. Water Piping Incl. Valves and Fittings	192,300	459,600
	.3 Water Treatment		
	.31 Chlorination Equipment	15,000	32,000
	.32 Chlorine Handling	3,500	4,000
	.4 Miscellaneous Other Unlisted Items Incl. Equipment Painting	10,000	39,000
	TOTAL ACCT. 232	$ 574,000	$ 1,441,850
233	CONDENSERS		
	.11 1 Pass Condenser Complete with Water Boxes, Expansion Joints, Extended Neck, Mfg. Supervision of Erection, Hogging Ejectors, Etc.	595,000	1,210,000
	.12 1 in. φ Aluminum Condenser Tubes	65,000	147,000
	.2 4 Condensate Pumps Complete with Motor Drives	95,000	270,500
	.3 Air and Gas Removal Equipment		
	.31 Dry Vacuum Pumps Incl. Motor Drives	12,500	24,600
	.32 Air and Gas Ejector Equipment Three (3) Stage, Twin Element with Inter and After Coolers	23,000	44,200
	TOTAL ACCT. 233	$ 790,500	$ 1,696,300
235	TURBINE PLANT BOARDS, INSTRUMENTS AND CONTROL		
	.1 Electrical Connections	24,000	26,000
	.2 Gage Boards, Instruments and Control	60,000	85,000
	TOTAL ACCT. 235	$ 84,000	$ 111,000

Table 7.4 — (Continued)

ACCT. NO.	DESCRIPTION	110 Gross MW_e TOTAL	325 Gross MW_e TOTAL
236	TURBINE PLANT PIPING INCL. INSULATION	Incl. in Acct. 228	
237	AUXILIARY EQUIPMENT FOR GENERATORS		
	.1 Excitation Panels, Switches, and Rheostats	Incl. in Acct. 231	
	.4 Fire Extinguishing Equipment		
	.41 CO_2 Fire Protection Equipment	30,000	43,000
	TOTAL ACCT. 237	$ 30,000	$ 43,000
	TOTAL ACCT. (23)	$	$15,959,190
24	ACCESSORY ELECTRIC EQUIPMENT		
241	SWITCHGEAR		
	.1 Generator Main and Neutral Circuits	8,000	9,000
	TOTAL ACCT. 241	$ 8,000	$ 9,000
242	SWITCHBOARDS		
	.1 Main Control Board	180,000	197,460
	.2 Auxiliary Power, Battery, and Signal Boards	441,000	515,240
	TOTAL ACCT. 242	$ 621,000	$ 712,700
243	PROTECTIVE EQUIPMENT		
	.1 General Station Grounding System	48,000	52,700
	TOTAL ACCT. 243	$ 48,000	$ 52,700
244	ELECTRICAL STRUCTURES		
	.1 Concrete Cable Tunnels	15,000	23,060
	.2 Cable Trays and Supports	88,000	133,800
	.4 Foundations-Electrical	5,000	5,380
	TOTAL ACCT. 244	$ 108,000	$ 162,240
245	CONDUIT WORK		
	.1 Conduit	110,000	143,800
	.2 Concrete Envelopes	17,000	21,715
	.3 Manhole and Covers	9,000	13,835
	TOTAL ACCT. 245	$ 136,000	$ 179,350
246	POWER AND CONTROL WIRING		
	.1 Main Power Cables (or Bus Duct)	124,000	204,000
	.2 Control, Auxiliary Power and Excitation Wiring	233,000	385,900
	TOTAL ACCT. 246	$ 357,000	$ 589,900

Table 7.4 — (Continued)

ACCT. NO.	DESCRIPTION	110 Gross MW_e TOTAL	325 Gross MW_e TOTAL
247	STATION SERVICE EQUIPMENT		
	.1 Station Service Transformers and Voltage Regulators	194,000	296,600
	.2 Batteries, Charging Equipment and Motor-Generator Sets	28,000	29,600
	.4 Remote Controls	—	—
	TOTAL ACCT. 247	$ 234,000	$ 340,200
	TOTAL ACCT. (24)	$ 1,512,000	$ 2,046,090
25	MISCELLANEOUS POWER PLANT EQUIPMENT		
251	CRANES AND HOISTING EQUIPMENT		
	.1 Turbine Room Crane	160,000	160,000
	.2 Other Cranes and Hoists		
	.21 Reactor Building Crane	59,000	59,000
	.22 Misc. Hoists and Trolleys	12,000	19,100
	TOTAL ACCT. 251	$ 231,000	$ 238,100
252	COMPRESSED AIR AND VACUUM CLEANING SYSTEMS		
	.1 Compressors and Accessories		
	.11 Station Air	6,000	13,800
	.12 Control Air	11,800	12,900
	.13 Station Air Receiver	1,150	1,175
	.14 Control Air Receiver	750	775
	.3 Vacuum Cleaning System	4,700	7,300
	TOTAL ACCT. 252	$ 24,400	$ 35,950
253	OTHER MISCELLANEOUS EQUIPMENT		
	.1 Local Communication, Signal, and Call Devices	26,000	33,600
	.2 Fire Extinguishing Equipment		
	.21 Fire Pump	12,600	12,600
	.22 Fire Pumps	6,800	13,300
	.23 Fire Extinguishers	1,200	1,800
	.3 Furniture and Fixtures	10,000	11,000
	.4 Lockers, Shelves, Cabinets	3,000	4,600
	.5 Cleaning Equipment		
	.6 Machine Tools and Other Station Maintenance Equipment	29,000	50,000
	.9 Diesel Generator Unit		
	.91 Diesel Generator	40,000	102,000
	.92 Diesel Oil Tank	400	800
	TOTAL ACCT. 253	$ 129,000	$ 229,700
	TOTAL ACCT. (25)	$ 384,400	$ 503,750

Table 7 5 — Nuclear Fuel Cycle Cost Data

Symbol	Description	Units	325 MW_e (8500 MW-d/tonne Burnup) Unit Value	Total Value	325 MW_e (7500 MW-d/tonne Burnup) Unit Value	Total Value	110 MW_e (6010 MW-d/tonne Burnup) Unit Value	Total Value
a	Total net power output	MW_e	318 9		318 9		103	
α	Average plant operating factor		0 80		0 80		0 80	
c	Thermal power output, per replacement batch[1] averaged over in-core residence	MW_t	279		279		91 3	
d	Average burnup	MW-d/MTU	8500		7500		6010	
e	Initial uranium loading, per replacement batch[1]	kg U	21,600		21,600		5558	
f	Final uranium loading, per replacement batch[1]	kg U	21,600		21,600		5558	
g	Initial U^{235} enrichment	w/o %	0 72		0 72		0 72	
h	Final U^{235} enrichment	w/o	0 14		0 17		0 23	
j	Final plutonium concentration	gm Pu/kg U	4 525		4 33		3 98	
k	Conversion charge UF_6-UO_2 or metal[2]	$/kg	—		61 25		—	
m	Fuel fabrication price[3]*	$	61 25 $/kg U	1,325,000		1,325,000	61 25 $/kg U	274,000
n	Total shipping and insurance charges, on new fuel and absorber sections[4]	$	2 $/kg U	43,200	2	43,200	2 $/kg U	8,950
β	Total recoverable scrap	%	—		—		—	
γ	Irrecoverable loss in making UO_2 or U metal	%	—		—		—	
δ	Irrecoverable loss in fabrication	%	2		2		2	
θ	UO_2 or metal fabrication time	mo	—		—		—	
λ	Core fabrication time	mo	6		6		6	
μ	New fuel shipping time, including storage on site	mo	0 5		0 5		0 5	
ψ	Spent fuel decay storage time, including shipping to processing plant	mo	4 5		4 5		4 5	
p	Total shipping and insurance charges, spent fuel[5]	$	5 $/kg U	108,000	5		5 $/kg U	22,375
q	Fertile material cost (if purchased)	$/kg	40 50		40 50		40 50	
r	Plutonium value	$/gm	12 00		12 00		12 00	
u	Control rod absorber sections, cost*	$	—		—		—	
w	Control rod absorber sections, lifetime	yr	—		—			
A	Value of uranium at initial enrichment, g	$/kg U	40 50		40 50		40 50	
B	Value of uranium at final enrichment, h	$/kg U	—		—		—	
C	Uranium burnup cost = A = B(f/e)	$/kg U	40 50		40 50		40 50	
D	Plutonium credit = r × j × f/e	$/kg U	54 30		51 96		47 76	
E	Average fuel throughput rate* = 365/12 × (c×α)/d × 1000	kg U/mo	795		901		369 6	
F	Chemical reprocessing batch size* (=f)	kg U	21,600		21,600		5558	
C	Chemical separation rate	kg/day	1000		1000		1000	
H	Chemical separation turn-around time[6]	days	8		8		8	
I	Total chemical reprocessing time = F/G + F/1000 + 35, if w/o $U^{235} \leq 5$ = F/G + F/150 + 35, if w/o $U^{235} > 5$	days	78 2		78 2		48 6	
J	Equivalent residence time at full power = (24×d×e/1000xc)	hr	15,720		13,870		15,960	
K	Spare fuel on hand* = E	kg U	795		901		369 6	
D_1	Pu^{239}	gms/kg U	3 325		3 25		3 1	
	Pu^{240}	gms/kg U	0 72		0 67		0 57	
	Pu^{241}	gms/kg U	0 48		0 41		0 31	
	Pu^{239}	$/kg U	39 90		39 00		37 60	
	Pu^{240}	$/kg U	8 64		8 04		6 84	
	Pu^{241}	$/kg U	5 76		4 92		3 72	
N	Cost of UF_6 to UO_2 or U-metal conversion = k × [1+(β+δ+γ/100)]	$/kg U	—		—		—	
P	Cost of fissionable material lost in conversion = A × γ/100 [1+(β+δ+γ/100)]	$/kg U	—		—		—	
Q	Use-charge during conversion =A/300 × θ × [1+(β+δ+γ/100)]	$/kg U	—		—		—	
R	Cost of fuel and absorber sections fabrication = [m+u(J/8760×w)]/e	$/kg U	61 25		61 25		61 25	
S	Cost of fissionable material lost in fabrication = A × δ/100[1+(β+δ/100)]	$/kg U	0 83		0 83		0 83	
T	Use-charge during fabrication = A/300 × λ × [1+(β+δ/100)]	$/kg U	0 83		0 83		0 83	
U	Cost of shipping new fuel and absorber sections = n/e	$/kg U	2 00		2 00		2 00	
W	Use-charge during shipping new fuel and storage on site = A/300 × μ	$/kg U	0 07		0 07		0 07	
X	Net burnup cost = C − D = A − f/e(B+r×j)	$/kg U	-13 80		-11 46		-7 26	
Y	Chemical separation cost = [16,400×(F/G+H)/F × f/e	$/kg U	22 5		22 47		40 13	
Z	Conversion of UNH to UF_6 = 5 60 × 0 987 × f/e	$/kg U	—		—		—	
AA	Conversion of PuNH to metal = 1 5 × 0 98 × j × f/e	$/kg U	6 65		6 50		5 85	
BB	Chemical processing losses = (0 013×B+0 02×12×j)f/e	$/kg U	1 09		1 04		0 96	
CC	Cost of shipping spent fuel to reprocessing plant = p/e	$/kg U	5 00				5 00	
DD	Use-charges during fuel decay storage, shipping to reprocessing plant and chemical reprocessing = B/300 × (ψ+I/30) × f/e	$/kg U	—		—		—	
EE	Use-charges during fuel in-core residence ≈ 1/300 × [A+B(f/e)/2] × e/E	$/kg U	1 85		1 63		1 02	
FF	Use-charges for spare fuel on hand = A/300 × K/e × 12	$/kg U	0 062		0 07		0 11	
GG	Fuel cycle costs at plant operating factor* = Σ (N through FF)	$/kg U	88 22		90 23		110 79	
HH	Annual fuel cycle costs at plant operating factor* = 12E × GG	$/yr	843,000		975,567		491,908	
II	Total annual fuel cycle costs at plant operating factor Σ(HH_i) 1 = number of replacement batches per core (see Note 1)	$		3,372,000		3,902,268		1,967,630
JJ	Unit energy cost = II/8760 × a × α	mills/kwh	1 51		1 746		2 726	

*Per replacement batch [1]

Each fueled lattice position contains two fuel elements arranged in tandem to permit more uniform burnup through repositioning during refueling. The total effective fuel length is 19.1 ft, which provides a total heat transfer surface area of 35,100 ft^2. The UO_2 fuel elements consist of 37-rod clusters. Each rod is composed of a column of cylindrical pellets, 0.5 in. in diameter sealed inside a 0.025-in. thick Zircaloy-2 tube.

To minimize neutron activation of the header room piping and equipment, neutron shield plugs are placed above and below the calandria. These shields are low alloy steel vessels filled with steel shot, through which water is circulated as a coolant. Stepped steel sleeves and grooved plugs reduce radiation streaming at each location where a pressure tube penetrates the shields.

Reactivity control, to perform the operation and safety functions, is accomplished by a total of 25 shim and regulating rods, each occupying a lattice position, and 25 safety rods. Shim and control rods are driven from below the core by electric motors. The safety rods are normally latched above the core and fall by gravity when released.

Primary Coolant System

The primary coolant system, together with its associated systems, is shown on the composite flow diagram, Fig. 7.1. D_2O at 499°F enters the bottom of the reactor and passes upward past the fuel elements, where it boils. A D_2O steam-water mixture at 14% quality and 515°F leaves the top of the core and flows through a piping distribution system to two steam drums. In the drums, the steam-water mixture is separated. The steam flows directly to the turbine-generator and the water, after mixing with feedwater from the turbine plant, is pumped to the reactor inlet. At full load, the turbine steam flow rate is 4.76×10^6 lb/hr and the recirculation flow rate is 34×10^6 lb/hr.

Primary coolant D_2O is recirculated by means of four horizontal pumps which operate in parallel. Each pump has a capacity of 20,000 gpm at a NDH of 200 ft. Control of the recirculation rate is effected by means of hydraulic couplings between the pump motor drives and the pump impellers. A constant core exit quality of 14% is maintained at all plant loads.

The steam, after passing through the turbine, is condensed in two main condensers. Two half-capacity condensate pumps are used to pump the D_2O condensate from the condenser hot well through the first three of four stages of extraction feedwater heaters. Feedwater pumps, located between the third and fourth stage of heaters are used to transfer the feedwater back to the steam drums for recirculation via the last stage of feedwater heating.

Extraction steam from the turbine flows to each of four stages of feedwater heating. The steam, after condensing in each heating stage, passes through integral drain coolers and is cascaded back to the first heating stage. At this point, all extraction heater condensate is pumped back into the condensate line to return to the reactor. The final feedwater return temperature is 387°F.

All pumps and valves used in the primary coolant system are equipped with seals designed to minimize and collect D_2O leakage.

Turbine Plant

The turbine plant, consisting of the turbine-generator, condensers, feedwater heaters, pumps, circulating water system, and associated auxiliaries, utilizes saturated D_2O steam from the reactor to produce a net electrical power output of a nominal 300 MW. Condensate from two condensers is returned to the reactor through two parallel banks of four stages of feedwater heating, with full-flow filtration being provided at the condensate pump discharge to remove particulate matter.

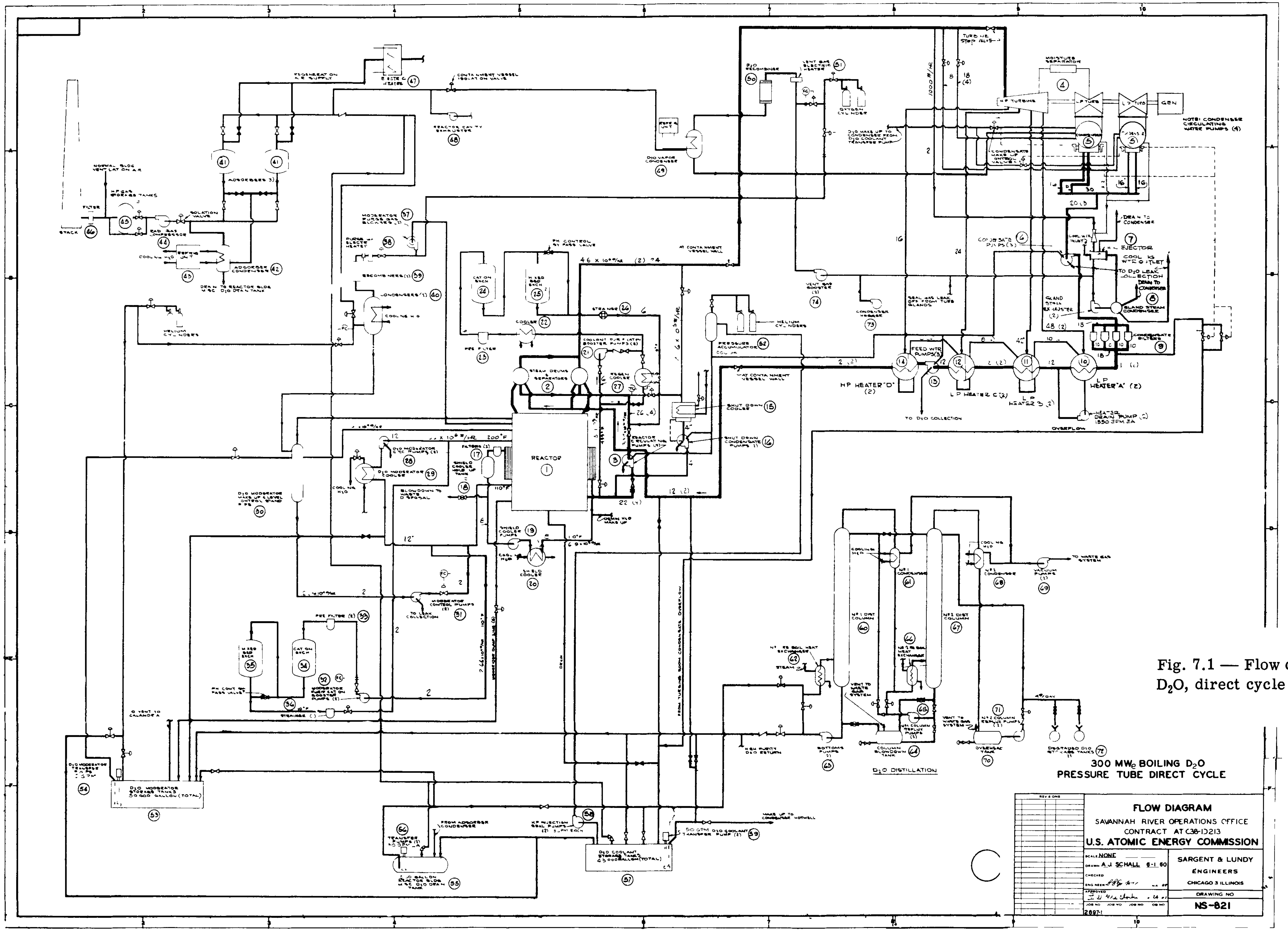


Fig. 7.1 — Flow diagram – 300 MW_e boiling D_2O, direct cycle plant (preliminary)

Reactor Building

Reactor containment is provided by a cylindrical steel vessel 135 ft in diameter and 190 ft in height from the top of the hemispherical dome to the bottom of the hemiellipsoidal base. The vessel is designed to contain an internal pressure of 31 psia, in accordance with applicable codes. The steel specified is SA-201, Grade B, carbon steel. General arrangement of the building and equipment is shown in Fig. 7.2. One inch of Flexcell separates the inside of the steel plate and the concrete shadow shielding. Steel plate for the cylindrical portion of the containment vessel and the bottom head is 0.97 in. thick; the hemispherical head is of 0.49 in. plate.

A 30,000 gal water tank is located in the upper dome to supply spray water in the event of a primary system rupture. If a rupture occurred, this water would be released through a spray system to lower the internal pressure and temperature.

Fuel Management

One of the primary reasons for developing D_2O-moderated reactors is to take advantage of the excellent neutron economy which permits operation on natural uranium fuel. This feature is of considerable interest to both domestic and foreign power producers. Maximizing fuel burnup and minimizing fuel fabrication costs of a natural uranium-fueled system will result in the lowest possible fuel cycle costs for any reactor system currently under development.

Studies were conducted in 1959 by S&L-NDA (see SL-1653) to investigate possible fuel management cycles for natural uranium-fueled systems. Refueling schemes considered were: batch, multizone batch, inward and outward radial shift, and outward radial shift with optional axial repositioning.

Fig. 7.3 shows the relative worth of these refueling schemes with respect to batch refueling. It has been calculated for the 300 MW_e plant that batch refueling would yield 3150 MW-d/metric ton.

Both SL-1815 and this report have used a four-zone outward radial shift with axial repositioning. This results in an average burnup of the discharged fuel of 8500 MW-d/metric ton for the 300 MW_e plant. However, due to a recent revision of the ground rules restricting the maximum allowable burnup to 8500 MW-d/metric ton, data are also shown for 7500 MW-d/metric ton average burnup.

Research and Development

Primary emphasis is being placed on physics, fuel element development, and component testing. Recent advances and proposed development in these areas that are applicable to this reactor are discussed in Section 5.

Economics

The capital investment and power cost for the boiling D_2O-cooled heavy water reactor is discussed in Section 4.

Potential Improvements

The plant and concept improvements and their effect on power generation costs are summarized in Section 6.

7.2 HEAVY WATER COMPONENTS TEST REACTOR (HWCTR)

E.I. du Pont de Nemours and Company

During the power reactor study conducted by the du Pont Company, it became apparent that there are no test reactor facilities available that can examine adequately a number of fuel elements of the length and at power densities required for economical D_2O-moderated reactors. The HWCTR is designed to fill this need.

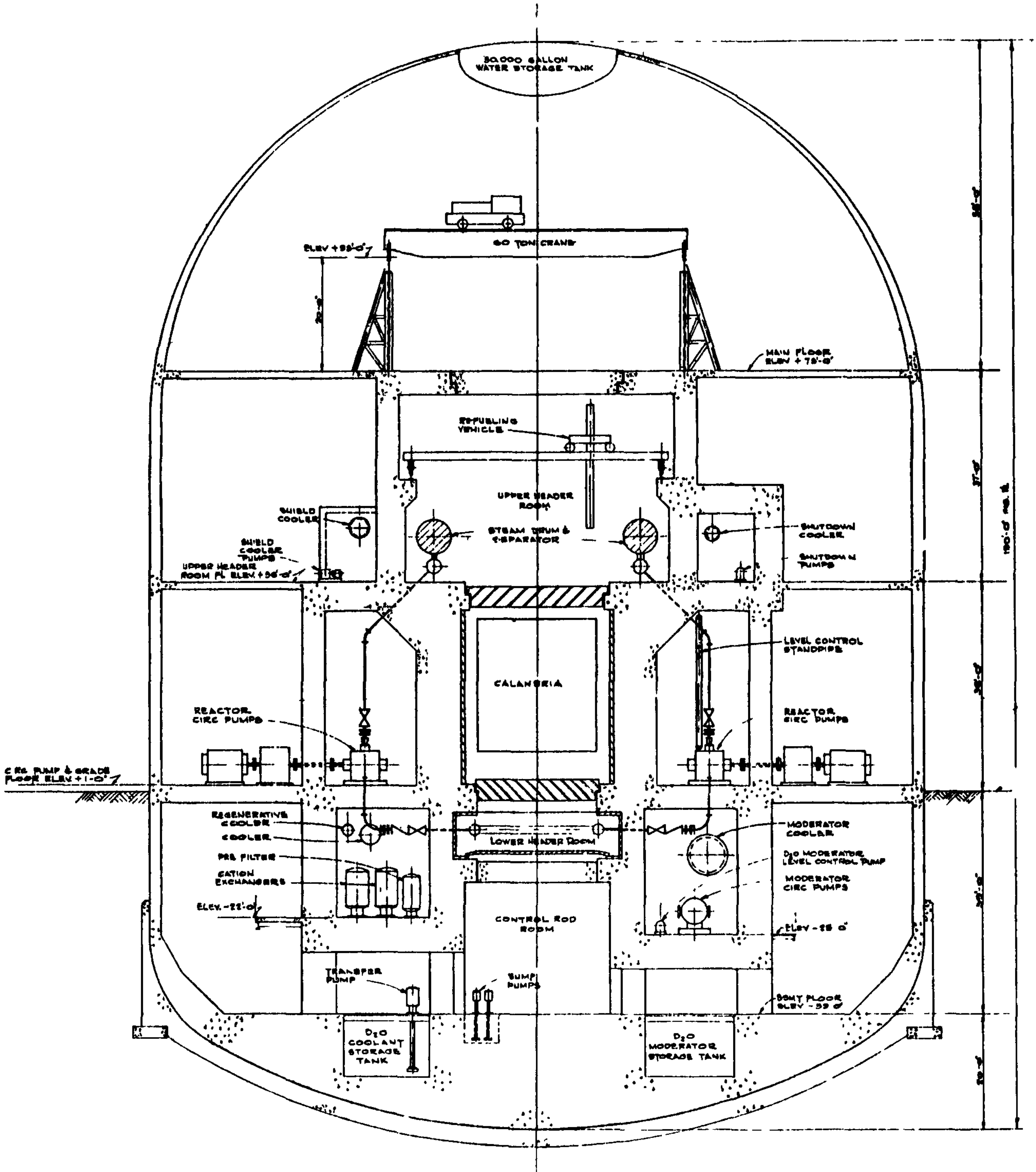


Fig. 7.2 — General cross section of reactor building – 300 MW_e boiling D_2O, pressure tube, direct cycle

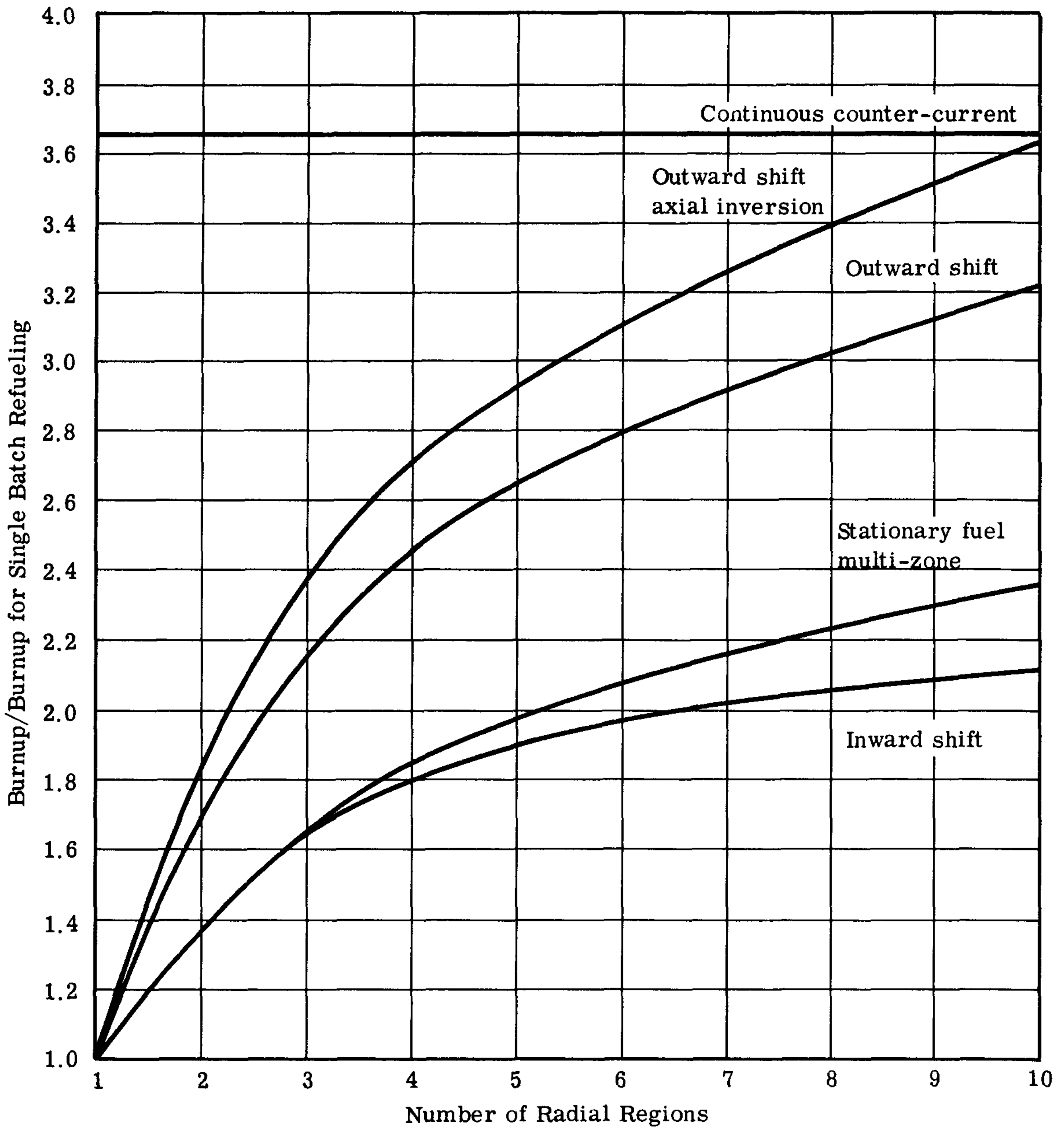


Fig. 7.3 — Attainable average burnup for multi-zone refueling schemes

Objective of the Project

The main objective of the HWCTR is the testing of heavy water-moderated power reactor fuel elements and the evaluation of power reactor components under power reactor operating conditions. Secondarily, the reactor may provide reactivity data as a function of burnup and fuel temperature.

Description of Concept

The HWCTR is a pressurized reactor that is cooled and moderated with heavy water. The reactor vessel arrangement is shown in Fig. 7.4. The reactor is designed to operate at a maximum pressure of 1500 psi, a maximum local D_2O temp rature of 550°F, and a maximum power of about 61 MW. Pressurization is by helium; there will be no boiling of the D_2O in the reactor. Plant parameters are given in Table 7.1.

The core consists of a central test region surrounded by a "driver" ring of enriched fuel assemblies. The test region assemblies will be candidate fuel assemblies of natural uranium.

The driver ring consists of 24 fuel tubes of oralloy-Zr alloy and contains a total of 25 kg of U^{235}. Target pieces containing boron as a burnable poison are placed along the axis of each assembly. A fuel burnup of 50% of the U^{235} is expected.

The reactor vessel is about 30 ft high and has a maximum ID of 7 ft. The average wall thickness is about 4 in. The bottom has 43 nozzles through which monitor pins for each fuel assembly and various instrument leads pass. The top head is bolted to the reactor and must be removed for charging fuel. The control and safety rod drive mechanisms are bolted to the head and lift with it when the head is removed.

Provisions are made in the reactor vessel for connecting six of the test positions to isolated coolant loops. Six inlet and six outlet nozzles can be connected to headers outside the tank wall (but inside the poured concrete shielding) so that three test positions can be connected to each of two isolated loops. Test assemblies, bayonet housing tubes, and connections to the nozzles can be inserted as required. It is anticipated that one of the isolated loops will be designed for boiling D_2O coolant and the other for liquid coolant. The boiling loop can be modified to handle gas coolant.

Heavy water flows into the top part of the reactor vessel at a rate of 9600 gpm. It flows down through the fuel elements, up through the moderator space, and out to two coolant loops. Heat is transferred to H_2O in the steam generators. The H_2O steam is vented to the atmosphere. The purification system keeps the moderator alkaline and keeps the dissolved oxygen and chloride contents within acceptable limits.

The HWCTR will be housed in a containment building about 70 ft in diameter by 125 ft high as shown in Fig. 7.5. The below-grade portion of the building, constructed of reinforced concrete with post-tension bands, has been erected and back-filled. The above-grade steel containment shell is being erected and system installation is under way.

Research and Development

Experimental evaluations of control and safety rod systems, flux distribution, and neutron economy have been made in the PDP and SE at room temperature and additional measurements are being made with D_2O temperatures up to 420°F in the PSE. Fuel element development of both driver and test elements is progressing and it is anticipated that a full loading will be available by the startup date, scheduled for the third quarter of 1961.

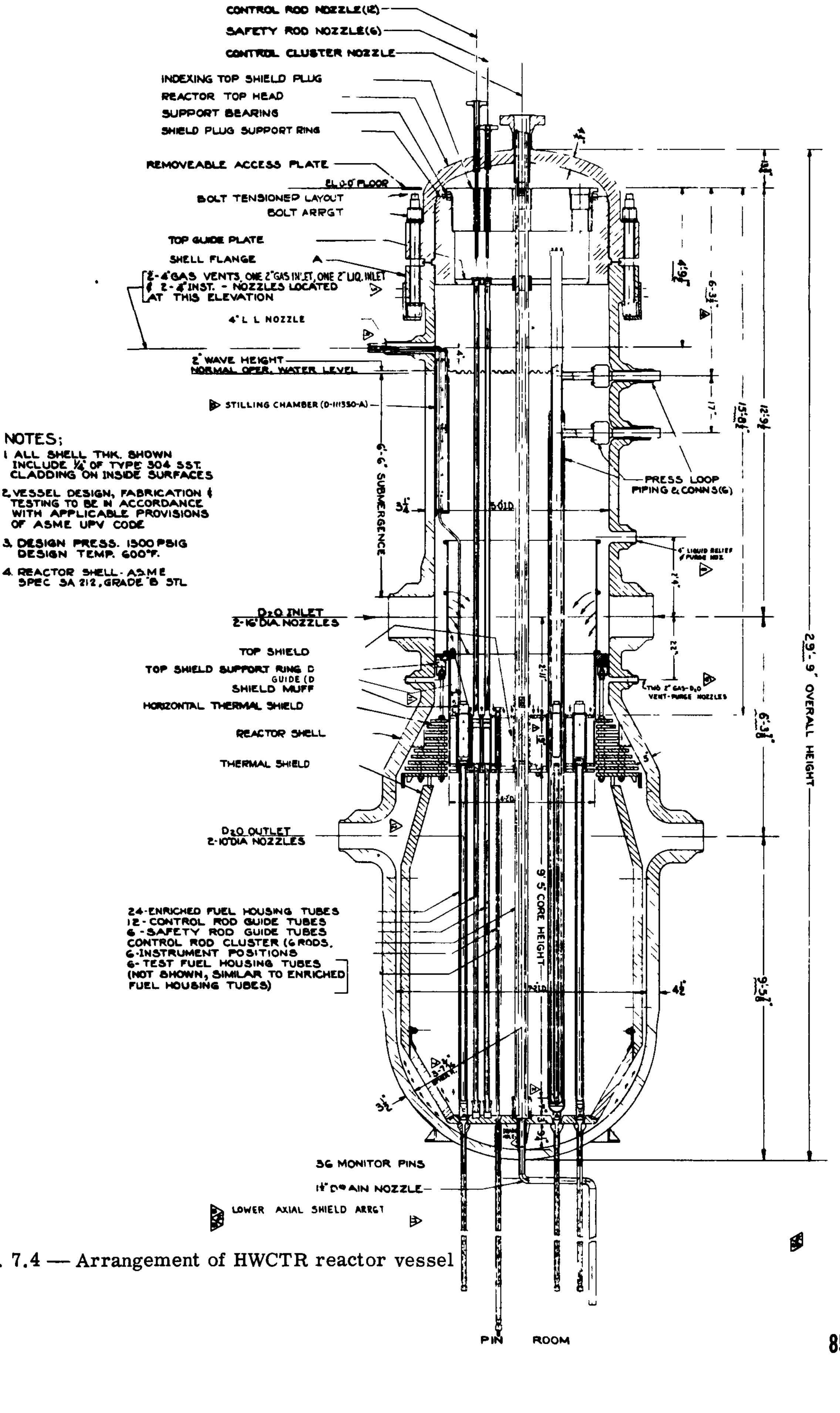


Fig. 7.4 — Arrangement of HWCTR reactor vessel

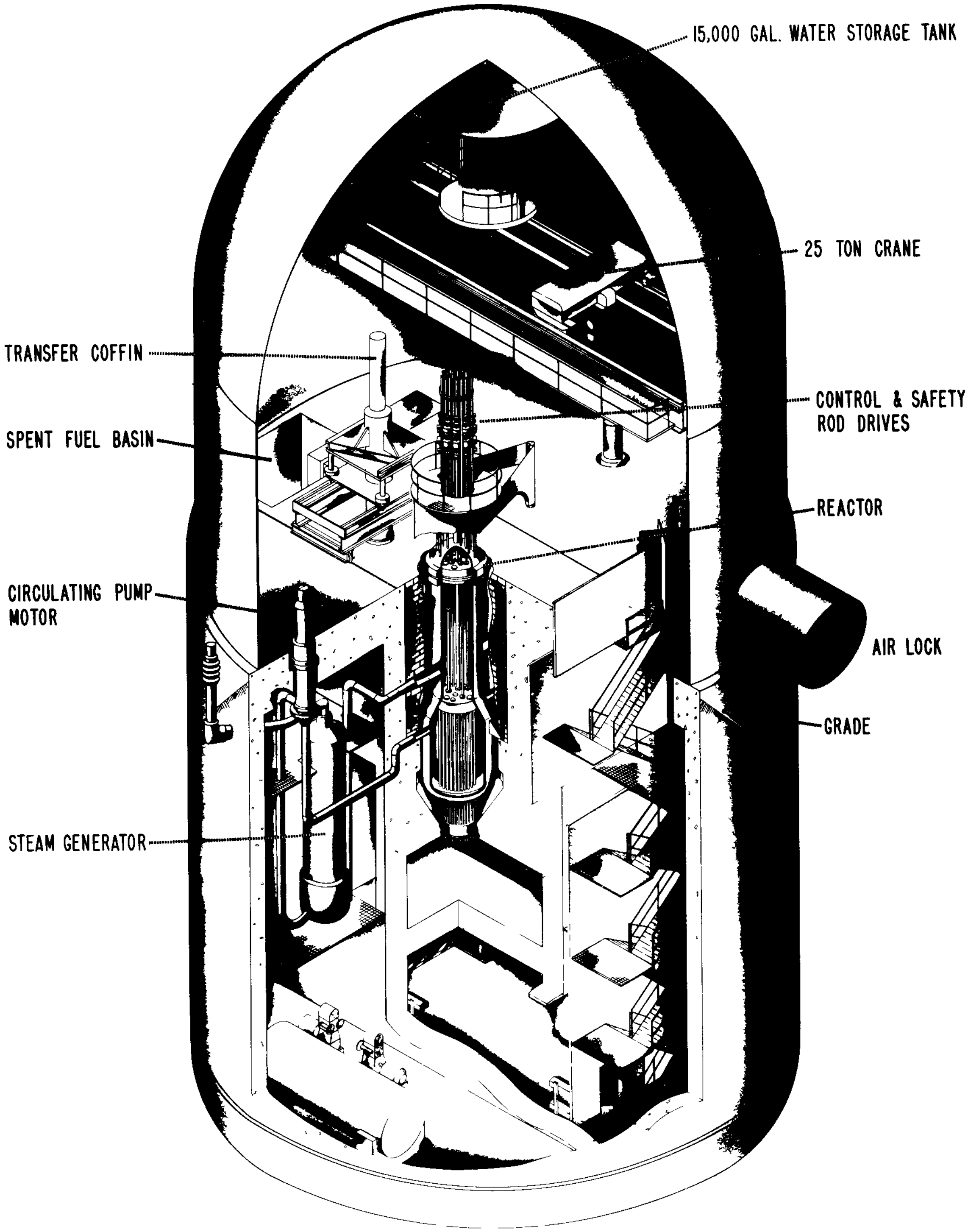


Fig. 7.5 — HWCTR – general arrangement

7.3 PLUTONIUM RECYCLE TEST REACTOR (PRTR)

Hanford Atomic Products Operation

General Electric

The Plutonium Recycle Program was started by the AEC in mid-1956. General Electric was chosen as the contractor responsible for determining the benefits of recycling the plutonium produced in a reactor as an enrichment type fuel. The PRTR is part of the overall Plutonium Recycle Program. It is not part of the D_2O-moderated power reactor program but is contributing substantially to the D_2O reactor technology.

Objective of the Project

The PRTR is an irradiation facility for the Plutonium Recycle Program. Its purpose is to provide data from which to determine the most economic scheme for recycling the plutonium produced in power reactors.

Description of the Concept

The Plutonium Recycle Test Reactor is a D_2O-moderated, pressurized D_2O-cooled, pressure tube-type reactor with a nominal power rating of 70 MW_{th}. The reactor arrangement is shown in Fig. 7.6. Plant parameters are given in Table 7.1.

The pressure tube design permits close control of the process variables in each tube, which allows considerably more experimental data to be obtained than could be in a pressure-vessel type reactor. Each of the 85 pressure tubes is, in effect, a test channel for fuel elements. Since a considerable part of the overall program deals with fuel element fabrication, the flexibility offered by the pressure tube concept is highly beneficial. Heavy water was selected as moderator and coolant for the reactor in order to obtain maximum flexibility as regards coolant channels and other neutron absorption aspects, as well as the proper lattice spacing to allow for the necessary process tube plumbing.

Since the objective of the Plutonium Recycle Program is to develop the technology required to utilize plutonium in power reactors, the PRTR will operate at conditions similar to those found in power reactors. Reactor coolant leaves the reactor at 530°F and 1050 psia, producing 425 psia steam in a heat exchanger. The moderator system is low pressure and operates at an average temperature of 143°F. Reactor control is by moderator level regulated by a gas balance system. Xenon poisoning and long-term reactivity are controlled by shim rods.

Research and Development

The PRTR incorporates most of the features considered to be typical of D_2O reactor designs, and is generally similar to most of the design and/or construction projects in the D_2O reactor program. Therefore, the contribution of the PRTR to D_2O-moderated reactor technology is of considerable interest to the program.

Probably the most important technical contribution was made in Zircaloy pressure tube fabrication and inspection techniques. The PRTR is designed so that pressure tubes, each about 20 ft long by $3\frac{1}{4}$ in. ID, can be removed easily. The operation program calls for periodic removal of tubes for destructive testing. In this way, the effect of irradiation on the properties of Zircaloy-2 will be determined.

Other items in the development program were the SS-Zircaloy joints and nozzle alignment during welding. The joint chosen, after extensive testing, was a simple flanged joint with a spirally wound gasket.

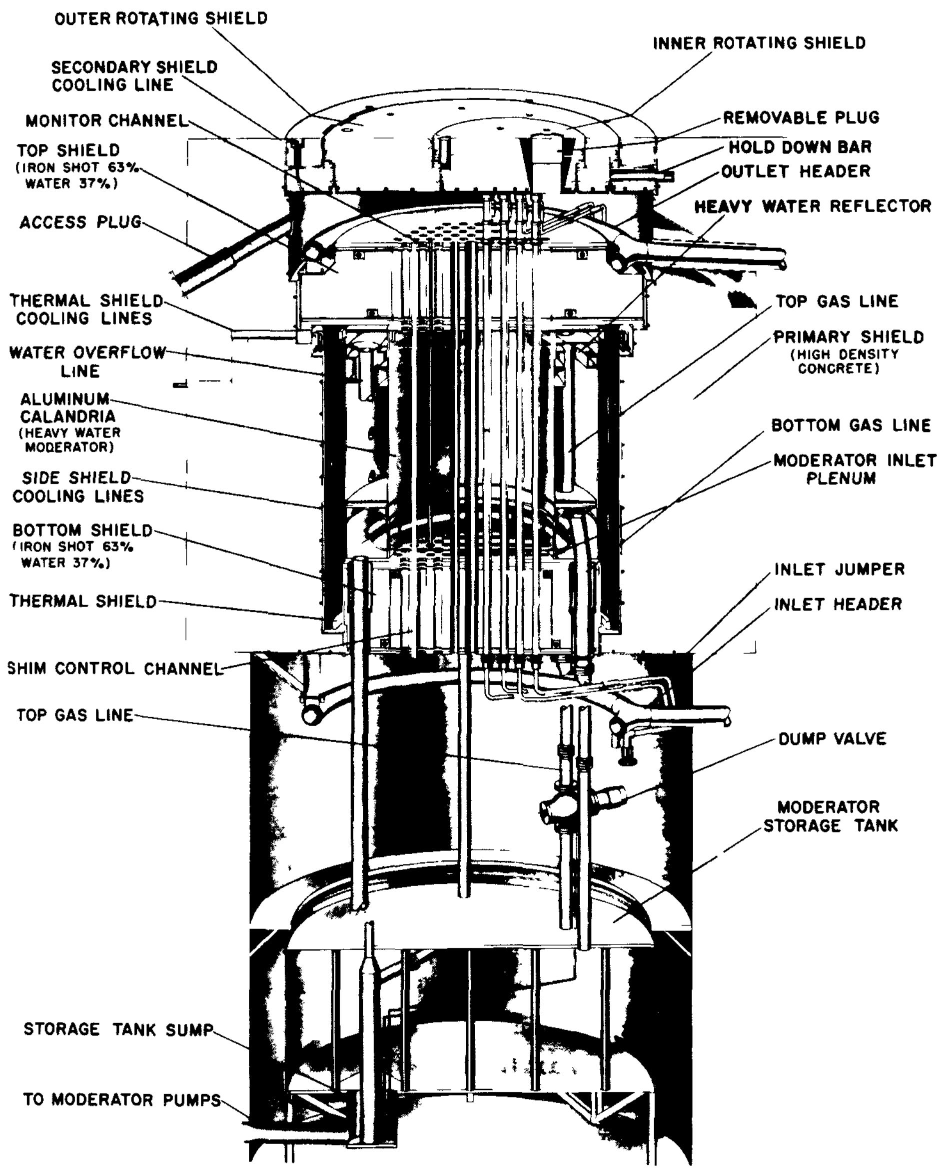


Fig. 7.6 — PRTR – reactor arrangement

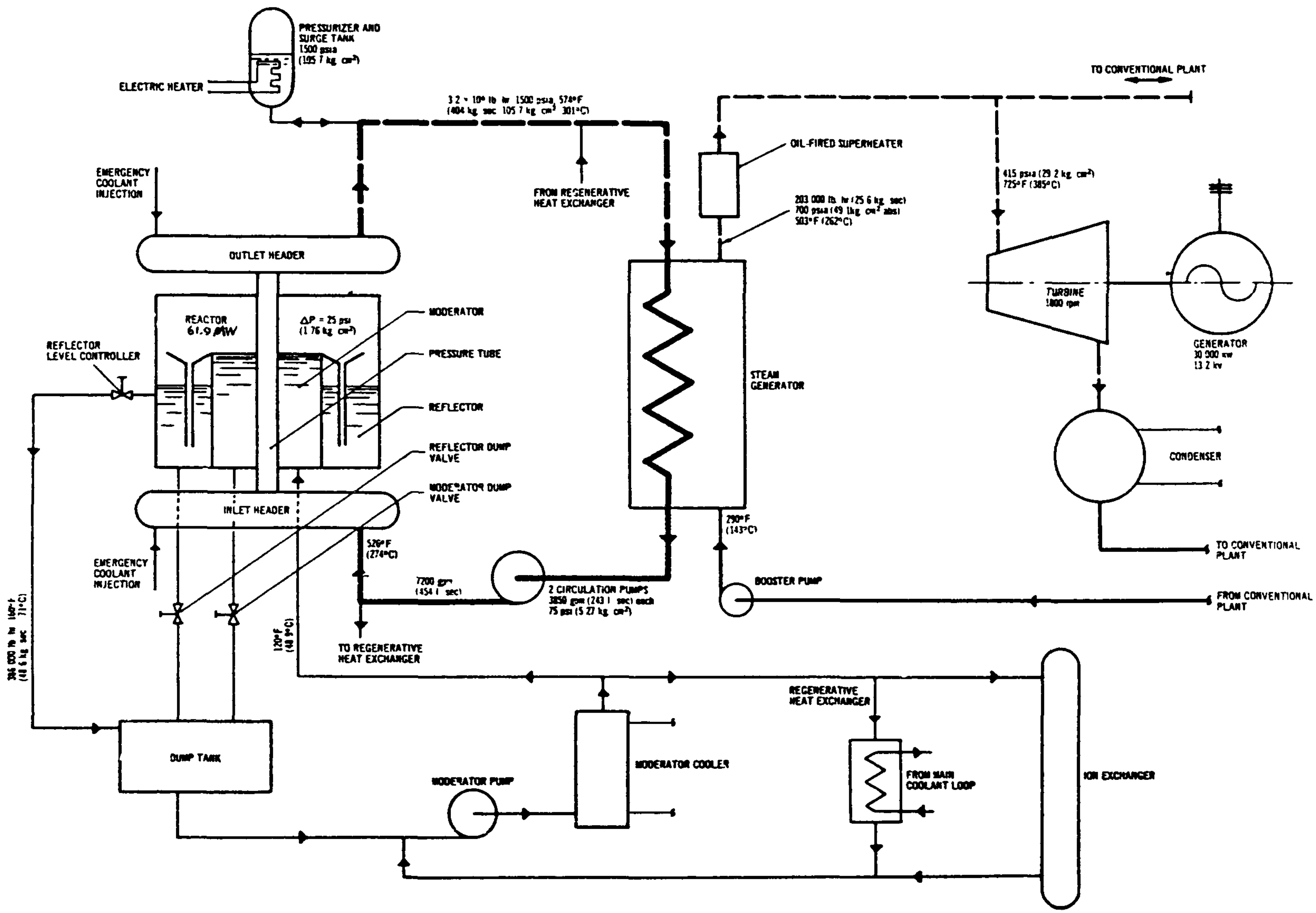


Fig. 7.7 — CVTR – flow diagram

Fuel element development at Hanford, although not specifically part of the PRTR project, is making substantial contribution to the state of the art. The initial loading of PRTR elements will be natural UO_2 swaged rods in 19-rod clusters with plutonium-aluminum spikes.

Schedule

The PRTR will be completed during 1960. Startup, including shakedown and approach to full power, is expected to take several months. Extensive "design," "critical," and "power" tests will be conducted during this period.

Potential Improvements

Improvements to the PRTR are not meaningful. Any improvements in fuel element performance which accrue from the program will be reflected in the use of these elements in power reactors of various types.

7.4 CAROLINAS-VIRGINIA TUBE REACTOR (CVTR)

Carolinas Virginia Nuclear Power Associates (CVNPA)

Westinghouse Atomic Power Department (WAPD)

Stone and Webster Engineering Corporation

CVNPA has contracted with Westinghouse under terms of AEC Contract AT(30-1)-2289 to conduct research and development work required to build a nuclear reactor plant. Stone and Webster have been retained as architect-engineers. A construction permit has recently been issued by the AEC.

Objective of the Project

The CVTR is a power demonstration prototype of a larger reactor which could operate on natural uranium. The reactor complex, including a fossil-fueled superheater, will be connected to an existing turbine-generator plant of the South Carolina Electric and Gas Company at Parr, S.C.

Description of the Concept

The reactor core has 84 fuel assemblies contained in 42 U tubes which are suspended in a moderator tank. Thirty two control rods are provided with drive mechanisms mounted above the moderator tank. Plant parameters are given on Table 7.1 and a flow diagram is shown on Fig. 7.7. The thermal output of the core is 61.9 MW gross, with 5.9 MW being lost to the moderator by direct convection and gamma heating.

The primary coolant is carried from the core through jumper tubes to an outlet header and a single primary coolant loop. The loop has valves for the isolation of the steam generator, a stop valve, a single vertical shell and tube type steam generator, a pressurizer, and two main coolant pumps. The mass flow rate of primary coolant is 3.3×10^6 lb/hr. The heat transferred to the secondary system is 191×10^6 Btu/hr.

Dry, saturated steam is produced in the steam generator at 605 psia and 487°F (full power conditions) at a rate of 202,000 lb/hr. This steam is then superheated in an auxiliary, oil fired superheater and fed to the steam header of the existing steam plant. The plant is rated at 17 MW_e net.

In addition to the major components mentioned above, the design also encompasses the normal complement of auxiliary systems found in plants of this type. These include charging and volume control, primary coolant purification, waste disposal, emergency injection, fuel handling, nuclear instrumentation, and others.

A steel-lined, concrete vapor container is also included for proper containment of the reactor and primary system and certain other components. A closed, recirculating ventilation system is provided with accommodation for removal and reclamation of heavy water vapor.

Auxiliary buildings are provided for housing of the various systems comprising the plant and furnishing necessary laboratory and administrative services.

The CVTR reactor core is approximately 83 in. in diameter and 96 in. high. The 42 U-shaped pressure tubes are $232\frac{3}{4}$ in. long and arranged in a rectangular lattice.

The pressure tube supports and surrounds the fuel assembly in the coolant stream. In the region surrounding each fuel element, the pressure tube is made of Zircaloy-4. The connecting U-bend is also Zircaloy-4, while the uppermost portion of the tube is AISI type-304 stainless steel. (AISI type-410 stainless steel may be substituted in part or wholly.)

The design criterion for the Zircaloy-4 portion of the pressure tube is an allowable stress equal to 1/3 of the minimum tensile strength at design temperature. The maximum average pressure tube temperature is calculated to be 275°F, and the design temperature is taken as 300°F. The design pressure is 1730 psi. The resulting wall thickness for the Zircaloy-4 in-core portion of the pressure tube is 0.253 in. for an ID of 3.53 in.

The straight portion of the pressure tube is fastened to the U-bend by means of a mechanical joint. The mechanical joint is used to minimize the number of unknowns in the design. The joint flanges are upset and machined for a Marman Conoseal closure. A pin is attached to the center portion of the U-bend to position the lower end of the pressure tube in a socket provided in the bottom of the moderator tank.

If means were not provided for limiting heat flow from the primary coolant to the moderator, the heat loss and attendant boiling of the moderator would be prohibitive for a pressure tube reactor. Thus, some form of thermal baffling must be provided, either inside or outside the pressure tube. In the CVTR design, an internal thermal baffle is used. In addition to limiting heat loss to the moderator, the internal baffle has the advantage of maintaining pressure tube temperature far below primary coolant temperature. Since the strength of any metal decreases as temperature increases, this baffle design permits the use of the relatively thin pressure tube wall previously described.

The CVTR thermal baffles provide stagnant insulating layers of primary coolant next to the pressure tube inner surface. These stagnant layers are formed by thin, concentric sleeves of Zircaloy. The Zircaloy thermal baffles consist of a U-section and two straight sections. Stagnant water layers are provided by four concentric sleeves. The inner sleeves on these lower baffles are on the order of 0.060 in. thick to insure adequate life, since these baffles must remain in place for the life of the U tubes. The intermediate and outer sleeves are made thinner (~0.020 in. thick) because they are protected from coolant flow by the inner sleeve. Dimples in the sleeves provide spacing between baffles as in the straight sections. The short transition baffles are locked to the pressure tube at the joint between the U-bend and the straight portions in each pressure tube leg. These baffles, in turn, hold the U-bend thermal baffle assembly in place by means of spring tabs at each end of the U-bend baffle.

Each fuel assembly consists of a 19-rod cluster of 0.500-in. OD fuel rods. The cluster is approximately 8 ft long. The fuel rods are hung from a grid attached to the upper end of the hex flow baffle. This allows independent axial expansion of each rod, thus eliminating thermal bowing. A grid attached to the bottom of the hex flow baffle serves to trap any fuel rod that might break loose from the top grid. All rods except the center one have a 0.100-in. high rib to provide spacing and to improve mixing of the coolant. Extending upward from the top grid is a thicker walled tube which is fastened to the pressure tube neutron shield plug by a remotely actuated latch.

The fuel assembly thermal baffles are sealed from the main coolant stream on the downstream end of each assembly and vented upstream so that coolant pressure drop through the assembly causes the hex flow baffle to squeeze in on the fuel rods and reduce any tendency for fuel rod flutter and fretting. Sealing is accomplished with piston rings at the lower end of the inlet-leg assembly and the upper end of the outlet-leg assembly. To minimize chances of jamming, these seal rings are designed to enter a close clearance only during the final inch of insertion of the fuel assembly.

A Marman Conoseal gasket is also used in the joint between the Zircaloy and stainless steel portions of the pressure tube. This joint is held together by a preloaded stainless steel sleeve which is welded to the stainless steel portion of the pressure tube. Differential expansion in the joint is reduced by extending the thermal baffles above the joint and locating the joint below the moderator surface. This results in cooler joint operation than if it were to be exposed to hot primary coolant on the inside and gas on the outside. However, the joint is designed to remain sound at operating conditions induced by broken thermal baffles, lowered moderator level, and other accident situations.

Immediately above the joint, the stainless steel tube ID increases approximately 1/8 in. so that the fuel assembly outer thermal baffle pulls free of the "tight fit" portion of the tube as soon as possible during refueling, lessening the possibility of a stuck fuel assembly. In the event that thermal baffles jam in a pressure tube during refueling, high force can be applied to break away a crimped joint which normally attaches the thermal baffles to the fuel assembly. Therefore, the fuel can still be removed from the reactor.

The upper end of the shield plug is attached to a cylinder which extends to a plug resting on a ledge in the upper end of the pressure tube. This portion of the pressure tube is termed the refueling port. Refueling will be accomplished by removing the U tubes to a spent fuel basin.

Coolant enters and leaves the pressure tube through the jumper connectors. These connectors are integral with the refueling ports and canted to the pressure tube plane in order to minimize the distance between pressure tube assemblies. The jumper connectors are fastened by bolts to the jumper support block. These bolts rest on sleeves which serve to reduce thermal stresses in the bolts. The entire weight of the U tube rests on this connection. This arrangement simplifies the problem of axial positioning of the pressure tube relative to the top neutron shield and the jumpers. Canopy welds seal the connectors to the jumper piping. Provisions are made for the insertion of an orifice cartridge in the inlet jumper connector to control flow to each pressure tube assembly.

A remotely actuated handling tool is capable of grasping either the neutron shield plug assembly or the fuel assembly, and can also remotely connect and disconnect them both.

Research and Development

The development program has concentrated on the U tube assembly and fuel element performance. Two loops and a fitting test facility have been used extensively.

Loop D is a high temperature, high pressure loop capable of subjecting the fuel element assembly to reactor operating conditions of temperature, pressure, and flow. Two phases of testing are being followed. Phase I consists of testing a stainless steel replica of the fuel assembly in order to determine the pressure drop and best arrangement of baffles to minimize the heat loss to the moderator. Phase IB will consist of a mockup including the top neutron shield, four thermal baffles, and other improvements. Phase II will be a proof test with fuel elements which are identical with those to be used in the reactor.

Loop E is a low pressure and temperature facility which is very flexible and can be used easily to test various fuel assemblies, orifices, neutron shield plugs, and other pressure tube internals. Tests have concentrated on the fuel element design and particularly the spacer wire geometry which gives the best compromise between coolant mixing and pressure drop.

A circulating autoclave facility is being used to test the various fittings in the U tube assembly. These include the Zircaloy-SS joint, the U-bend joint, and the refueling port.

In addition to the work above and the analytical work which will continue, other phases of the work which are just starting or will be started in the near future are: (1) in-pile loop experiments, (2) critical experiments, (3) moderator flow tests, and (4) miscellaneous programs.

The in-pile loop will test a one-half scale pressure tube with a cluster of seven fuel rods, complete with thermal baffles. Coolant will be pumped at 1650 psi and 550°F. The fuel assembly and pressure tube will be examined in a hot cell at the completion of the test. The in-pile loop tests will be augmented by capsule irradiations.

Critical experiments started in the first half of 1960. Both half and full scale cores will be investigated. Primary emphasis is being placed on determining the effect of loss of coolant on reactivity.

A one-fifth scale model of the moderator tank is being used to determine the flow pattern and make design changes as necessary to assure proper cooling without stagnant moderator zones.

In addition to the above main tasks, several small tests will be run. Small scale models of the pressure tubes will be used to determine the coolant flow stability. The thermal resistance between Zircaloy and stainless steel will be measured. Ceramic insulation for the pressure tube will be investigated as an alternate thermal baffle.

Economics

The CVTR, being a power demonstration prototype with fossil fuel superheat and utilizing an existing steam plant, cannot be used to show the economics of a full scale power station. The cost of power is not a meaningful number. Available cost data are presented in Tables 7.6 and 7.7.

Potential Improvements

The main improvements which can be incorporated in the CVTR concept are the same as for other pressurized reactors. The U tube construction will permit pressures as high as desired, as new fuels and materials are developed. The core construction, U tubes suspended in a moderator tank, completely eliminates the restriction on core size. The main power cost reduction will come from increased plant capacity.

7.5 ECNG-FWCNG GAS-COOLED REACTOR PROJECT

Florida West Coast Nuclear Group (FWCNG)

East Central Nuclear Group (ECNG)

General Nuclear Engineering Corporation (GNEC)

The Florida West Coast Nuclear Group (FWCNG) was formed by Tampa Electric Company and Florida Power Corporation. FWCNG has proposed, subject to the considerations of AEC Contract AT(38-1)-200, to construct and operate a 50 MW_e nuclear power plant integrated with their electric systems. The plant site is located in Polk County, Florida, approximately 35 miles east of Tampa, Florida.

Research and development work necessary, before final design, construction, and operation can be undertaken, is being conducted jointly by the East Central Nuclear Group, the United States Atomic Energy Commission, and the Florida West Coast Nuclear Group. General Nuclear Engi-

Table 7.6 — Estimated Capital Cost* – Prototype Nuclear Power Plant – Carolinas-Virginia Nuclear Power Associates, Inc.

Generating Facilities	Cost, $
Land and land rights	
Land and privilege acquisition	46,200
Structures and improvements	
General yard improvements	212,400
Auxiliary building	302,800
Control bay	109,000
Office and service building	228,000
Spent fuel building	292,300
Fire pump house	4,700
Site laboratory	71,500
Weather tower	33,500
Vapor container structure	1,814,600
Reactor plant equipment†	
Reactor equipment	3,274,700
Primary loop system	3,042,400
Auxiliary systems	3,359,200
Steam and feedwater systems	320,500
Service systems	210,700
Miscellaneous reactor plant equipment	77,900
Preliminary operation and tests	127,200
Steam superheater and fuel oil facilities	263,000
Turbine generators‡	—
Accessory electric equipment	916,200
Miscellaneous power plant equipment	289,800
Allowance for price adjustments	
Material price adjustments	411,000
Wage adjustments	205,500
Engineering investigations, surveys, and reports	46,700
Carolinas-Virginia Nuclear Power Associates, Inc. charges	
Administrative cost	1,715,000
Financing cost	400,000
Interest during construction	1,444,500
Total generating facilities	19,319,300
Transmission and distribution facilities‡	
Startup Costs	180,700
Total capital cost	19,400,000

* Including allowances for price adjustments on material and labor.

† Excluding cost of nuclear fuel and fuel fabrication.

‡ It is planned to install the proposed reactor at an existing steam-electric generating station having ample facilities for electric power generation, transmission and distribution. Therefore, expenditure for these items will not be required.

Table 7.7 — Estimated Postconstruction Operating Expenses – Power Demonstration Reactor – Carolinas-Virginia Nuclear Power Associates, Inc.

	Annual Steady State Operating Cost for 19 MW_e (Gross) Prototype Plant, $
Revenue	
Sales of steam	
Operating Costs	
Reactor and steam generator system	
Nuclear fuel cost	
Fuel fabrication and assembly	636,900
Material losses and scrap UO_2 reprocessing	8,860
Burnup, U^{235}	182,200
Reprocessing of spent fuel elements, including shipment and cask rental	98,600
Use charge	16,050
Shipping and shipping containers (new fuel)	8,110
Core capital charge	67,800
Gross nuclear fuel cost	1,018,520
Credit for plutonium at $12/g	-68,300
Net nuclear fuel cost	950,220
Heavy Water Rental at 4%/yr	108,000
Heavy Water Losses at 3%/yr	81,000
Heavy Water Insurance at 1%/yr	27,000
Heavy Water Purification at 1%/yr	27,000
Fuel Oil at 7.23¢/gal	144,600
Electricity at 6 mills/kwhr	82,700
Steam at 24¢/1,000 lb	2,600
Supervision	92,000
Operating Labor	157,000
Maintenance Labor	46,000
Materials and Services	95,700
Insurance	75,000
Third Party Liability Insurance	90,000
Net Earnings after Taxes (6%)	—
Taxes (State and Local)	15,000
Rental of Property	10,000
Gross Operating Costs to CVNPA	2,003,820
Less net steam sales payments and waiver of use charges by AEC*	-240,000
Net Operating Cost to CVNPA	1,763,820

*SROO estimate.

neering Corporation is the Nuclear Project Engineer, and American Electric Power Service Corporation is the Principal Design Engineer. The project schedule has been oʂ ɔriented to permit the use of beryllium fuel cladding in the first core.

Objectives

The 50 MW_e prototype is intended to demonstrate the gas-cooled, heavy water-moderated power reactor.

Concept Description

The FWCNG Nuclear Power Plant utilizes an advanced gas-cooled, heavy water-moderated, pressure tube reactor which employs slightly enriched uranium dioxide as the nuclear fuel. A perspective of the reactor is given in Fig. 7.8. The reactor has a net electrical output of 50 MW and is a prototype of a larger reactor (300 MW) which will be capable of operating with natural uranium as fuel. The primary coolant of the reactor is carbon dioxide and is contained at a nominal operating pressure of 500 psi. The coolant enters the reactor at 550°F and leaves at 1050°F. The coolant leaves the reactor through two separate loops and enters two steam generators where it transfers its energy to the steam system; it is then circulated back to the reactor. Each loop is complete with its own heat exchanger, valves, and blowers. About 9% of the total reactor power appears in the D_2O moderator and is removed by two moderator heat exchangers wherein this energy is used to improve cycle efficiency by heating condensate. The initial concept proposed a single pressure-reheat steam cycle with pressures of 1465 psia/215 psia and with temperatures of 950°F/950°F. A preliminary heat balance is shown in Fig. 7.9. Recent optimization studies favor a nonreheat cycle at substantially the same upper pressure and temperature.

A steel, cylindrical, containment vessel houses the reactor and its CO_2 coolant system. The steam plant is of conventional design, is not radioactive, and is located within a turbine building outside the containment vessel. A spent-fuel storage building is located adjacent to the reactor enclosure. The plant's central control room is located in the heater bay of the turbine building. An office and service building, which is attached to the turbine building, houses the offices, machine shop, chemistry laboratory, and storerooms for the plant. Other plant structures include a gatehouse, a circulating water intake structure, and a switch-yard.

A tabular summary of plant data is presented in Table 7.1.

Research and Development

Extensive research, development, and testing programs are in progress to provide a sound design basis for key components and to demonstrate their reliability. All components whose application is not proven will be proof-tested where plant safety and reliability are affected. The most significant portions of the program are briefly discussed.

In addition to the work conducted or sponsored by this project, results of relevant work on other projects are being closely followed and considered. In particular, there is very close coordination between this project and the gas-cooled project at ORNL. Both the Canadian work on heavy water reactors and the European gas-cooled experience are being utilized wherever such work and experience are pertinent to this project.

1. Fuel Development Program

Critical areas, where safety margin may be sensitive to design and operating conditions, have been identified through analysis. Development and testing activities are now proceeding in these areas.

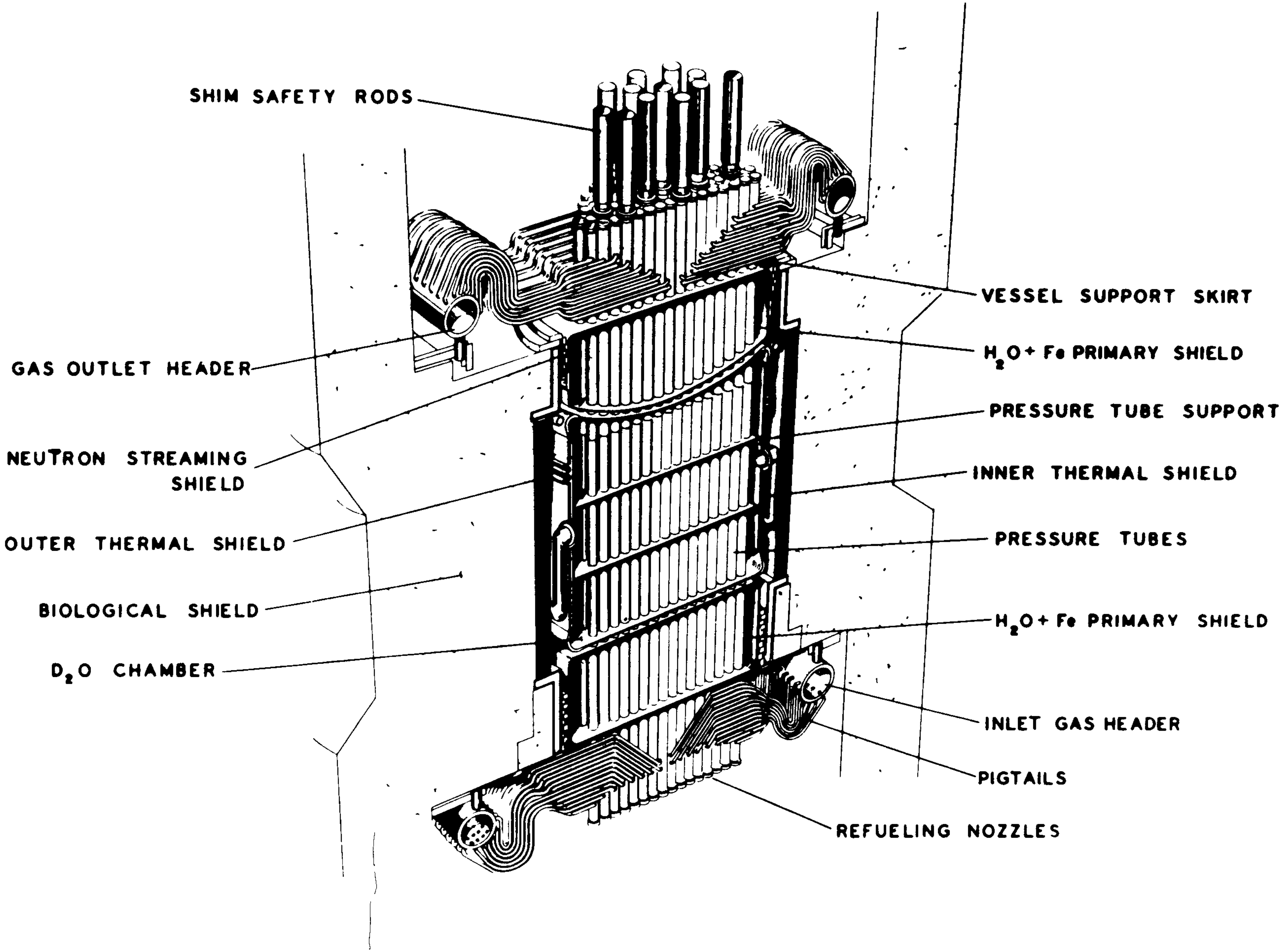
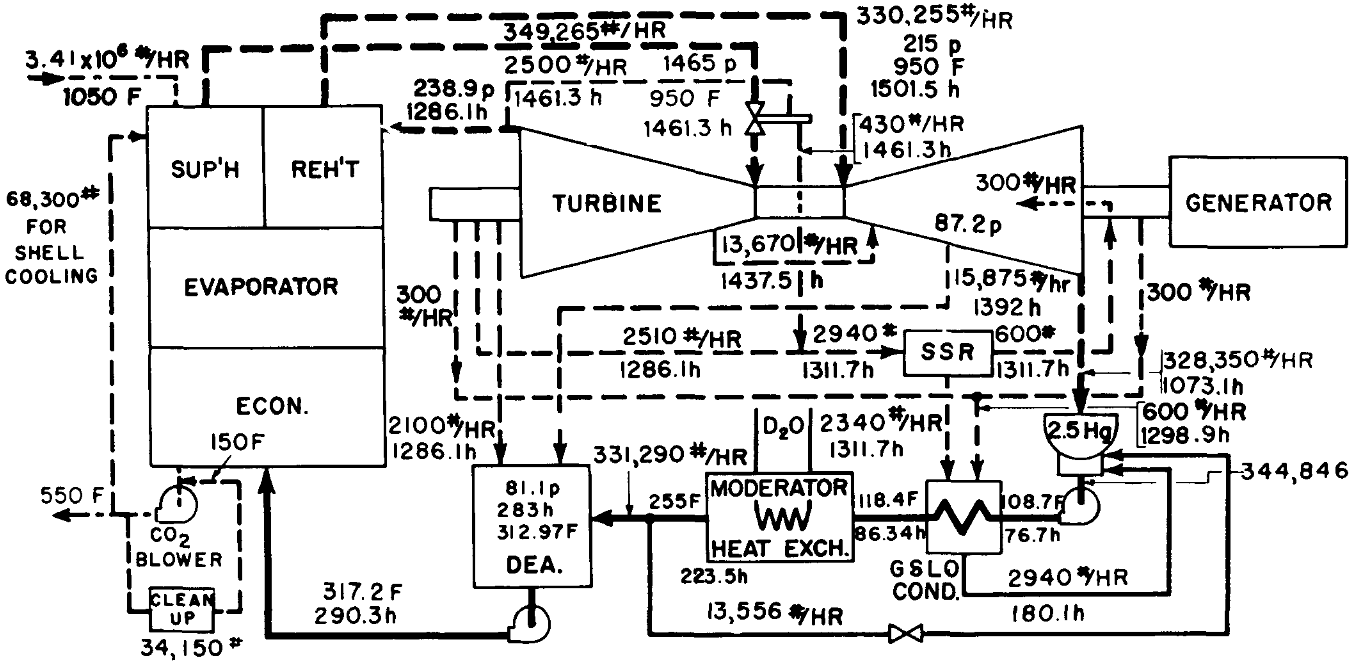


Fig. 7.8 — FWCNG – reactor arrangement



REACTOR OUTPUT	151,626 KW
MODERATOR HEAT LOSS (9%)	13,646 KW
REACTOR NEUTRON SHIELD HEAT LOSS(0.6%)	910 KW
REACTOR HEAT INPUT TO STEAM GEN.	467,785,780 BTU/HR
TOTAL HEAT INPUT TO STEAM GEN.	479,663,770 BTU/HR
STEAM GENERATOR APPROACH TEMP.	50 F

GROSS GENERATION	57,228 KW
MISC. AUX. POWER	1500 KW
FEED PUMP POWER	786 KW
BLOWER POWER	4730 KW
NET GENERATION	50,212 KW
NET HEAT RATE	10,306 BTU/KWH

Fig. 7.9 — FWCNG gas-cooled reactor – flow diagram (preliminary)

a. Heat Transfer and Fuel Bundle Testing

Heat transfer tests are being performed at conditions which closely simulate reactor operating conditions. The location and magnitude of hot-spot factors of various origins used for design calculations will be partially checked by these tests. Heat transfer performance will be determined for multiple rod bundles, simulating those to be used in the reactor. Mockups of fuel bundle assemblies are also being subjected to structural testing, both short and long term. The expected temperature gradients are introduced and actual deflections are measured. These are correlated with stress analyses and related back to measurements from the heat transfer tests. Vibration and flow tests are also being conducted.

b. Materials

As a result of the reoriented R&D program, beryllium has been tentatively selected as the clad material. Testing is continuing to determine the long term behavior of this and other materials under the corrosion conditions expected in the reactor. Tests will also be conducted to determine the effect of introducing moisture into the main CO_2 loop, a condition which would result from a leak in the steam generator. An in-pile test will be run to determine the corrosion effects of CO_2 radiolytic decomposition products on the selected material. Creep tests on the material selected for the fuel clad will be run in a CO_2 environment.

c. Mechanical and Fabrication Process Development

Fabrication process development of fuel rods and bundles is being carried out to develop a bundle of high reliability. Bundles produced by this program will be subjected to extensive tests including in-pile testing at maximum operating conditions prior to fabrication of the first fuel load, thereby permitting improvements to be incorporated.

d. Release of Fission Gas

Oak Ridge National Laboratory is cooperating in a comprehensive program to measure, among other parameters, the release of fission gas from UO_2 fuel rods which are operated at high-performance reactor conditions. The results from these tests will be the basis for final fuel design for this project. The tests will also determine the relative performance of cored pellets. UO_2 performance data from the PWR project are also being considered in the design.

e. In-Pile Loop Test of Complete Fuel Bundle Assembly

Most of the problems associated with the fuel design will have been evaluated through the tests and analyses mentioned above. A final proof of the performance of the bundle design will be obtained by an in-pile loop test which subjects the fuel bundle to operating conditions similar to those experienced at full power operation. This loop is now being fabricated, and at least two years of testing have been planned.

2. Zircaloy-2 Pressure Tubes and Flow Liner

Four, 20-ft long, welded Zircaloy-2 pressure tubes have been fabricated. Tensile and burst tests on specimens taken from these tubes have exceeded specifications. A similar fabrication and testing program will be undertaken for seamless extruded Zr-2 pressure tubes.

A second phase of this program will determine the effects of a pressure tube failure (carrying 500 psi CO_2) on the reactor vessel and other pressure tubes and will aid in locating and designing a relief diaphragm to prevent rupture of the reactor vessel. This test will be performed by utilizing a quarterscale model of the prototype reactor core in which tubes will be purposely made defective so that they fail at a pressure approximately equal to reactor pressure.

The pressure tube is maintained at 300°F or less by contact with the relatively cool D_2O. The coolant gas temperature ranges from 550°F to 1100°F. A thermal barrier must be provided to limit heat transfer to the moderator. Models of ceramic liners have been tested and metallic liners are also being investigated.

a. Transition Joint

Intensive research into dissimilar metal joints indicates that a reliable and simple joint to use between the Zircaloy-2 pressure tube and the stainless steel stub tube is the intermediate-material type. Nickel-iron will be the intermediate material, and it will be rolled and brazed to the Zircaloy-2 and will be welded to the stainless steel.

The joint between the Zircaloy-2 pressure tube and the stainless steel stub tube will be tested at reactor conditions of temperature and pressure and will be subjected to thermal cycling and vibration.

b. End Plug

The stub tube end plugs must combine reliability and leaktightness. Emphasis will be placed on the use of known and proven seals and backup components; the end plugs will be tested under conditions similar to those existing during reactor operation.

3. Reactor Inlet and Outlet CO_2 Gas Assemblies

All components of the manifold assembly are being tested individually and also, in final test, as a complete assembly. With the change in cladding from stainless steel to beryllium the lattice spacing increased thus simplifying access to the pressure tubes. A pigtail design has now replaced the plenum box and bellows design.

4. Fuel-Handling Machine

A model of the on-power refueling machine will be built and will be extensively tested using the fuel bundle, flow liner, and stub tube end plug designs which are now being developed.

5. Control Rod and Drive

A full scale model will be fabricated for testing.

Schedule

The project is being reoriented and no definite dates have been set.

Economics

Two plant sizes have been investigated; a 300 MW_e plant and a 50 MW_e prototype plant. The latter plant, using enriched fuel, is estimated to require a capital investment of $29,391,000 excluding land and D_2O. The 300 MW_e plant has been studied with natural uranium fuel as well as three enrichments, all in beryllium cladding. Capital costs and operating expenses for the 300 MW_e plant are given in Tables 7.8 and 7.9. It is interesting to note that the fuel cycle cost is cut to less than 1/2 by increasing the enrichment from natural to 1.15 atom %. This is the direct result of increasing the average fuel burnup from 6000 to 19,600 MW-d/ton.

Table 7.8 — FWCNG Gas-Cooled Reactor – Estimated Capital Cost (Reference 52) for 300 MW_e Full Scale Plant

March 1, 1960

Capital Cost Item	Cost, $
Land	360,000
Site work	910,000
Buildings and structures	8,333,000
Reactor plant	27,445,000
Turbo-generator system	12,399,000
Accessory electrical equipment	3,218,000
Miscellaneous power plant equipment	624,000
Total direct costs	53,289,000
Engineering services including general and administrative	10,000,000
Contingency	6,300,000
Startup	500,000
Escalation	8,300,000
Interest during construction (42 months including six-month start-up period)	8,590,000
Total indirect	33,690,000
Total capital cost	86,979,000
Capital cost, $/kw	290

Table 7.9 — FWCNG Gas-Cooled Reactor Electric Generating Costs (Reference 52) 300 MW_e Full Scale Plant

March 1, 1960

	Cost, mills/kwh				
Fuel Enrichment, percent	Natural	0.83		1.00	1.15
Fixed Charges (14%, 80% capacity factor)			5.8		
Heavy-water Inventory (12.5% per year)			0.79		
Fuel-cycle Costs	2.6	1.8		1.37	1.18
Operation and Maintenance			0.93		
Heavy-water makeup (1% Losses)			0.06		
Total Energy Cost	10.18	9.38		8.95	8.76

Note: All cost figures based on 80% capacity factor.

7.6 NUCLEAR POWER DEMONSTRATION REACTOR (NPD-2)

Atomic Energy of Canada Limited (AECL) Hydro Electric Power Commission of Ontario

Canadian General Electric Company Limited

The Canadian heavy water reactor program was initiated as a wartime measure and gave rise to three experimental facilities: (1) ZEEP in 1945, (2) NRX in 1947, and (3) NRU in 1957. In 1955 a study of a 10 to 20 MW electrical station was started and had progressed into the construction stage of NPD-1 by 1957. At this time a study of large electric plants revealed that a number of changes in design of NPD-1 should be made in order to make it represent a prototype of the large plant. Construction of NPD-1 was stopped while the design change-over to NPD-2 was made and was renewed in 1958. NPD-2 is scheduled to operate early in 1961.

Objectives

NPD-2 is an electric power generation station with two objectives: (1) to produce electrical power, and (2) to serve as a power demonstration prototype for a larger plant, CANDU.

Concept Description

The NPD-2 is a pressurized D_2O-cooled, D_2O-moderated reactor plant as illustrated in Fig. 7.10. It is fueled with natural UO_2 in Zircaloy cladding. The reactor vessel, or calandria, consists of a horizontal aluminum barrel-shaped shell with 132 aluminum tubes for fuel positions spaced on a 10.5 in. square pitch. It is surrounded by a cylindrical tank forming a $13\frac{1}{2}$ in. minimum annulus for a light water neutron reflector and shield. The core tank is 12 ft 7 in. long and 14 ft 8 in. maximum diameter which tapers to 12 ft 0 in. at the heads. This barrel shape provides a maximum of 21.5 in. D_2O reflector around the core at the midplane.

Moderator level is controlled by a helium pressure balance system by means of a weir at the bottom of the core tank. Emergency shutdown of the reactor can be made by release of the helium pressure, permitting the D_2O to flow to the dump tank through three 24 in. diameter pipes.

Each fuel position consists of a 4 in. ID by 0.052 in. wall aluminum calandria tube and a 3.25 in. ID by 0.163 in. Zircaloy-2 pressure tube with appropriate end fittings. The pressure tube end closures have provisions for the refueling machine to make a pressure tight seal prior to removing the diaphragm type closure plug. On-power refueling will be used from each end of the reactor, providing continuous countercurrent refueling.

Two types of fuel elements are being considered: clusters of seven 1 in. rods and $19\frac{1}{2}$ in. rods. Both elements fit into the same size tubes. The 19-rod elements may be used in the high flux section of the reactor if central fuel temperature limits the UO_2 pellet diameter.

Reactor control requirements are minimized by the use of on-power refueling which permits the reactor to operate under normal conditions with only 0.05% excess reactivity. Moderator level and temperature are used as the normal control element plus an enriched booster rod which can be inserted slowly to override the after-shutdown poison buildup.

The reactor is connected through heat exchangers and a steam drum to a 20 MW_e turbine. Plant characteristics are summarized in Table 7.1.

Research and Development

The Canadian research and development program embraces virtually every phase of nuclear reactor technology ranging from advanced concept studies through full scale reactor plant construction. The main parts of the current program include:

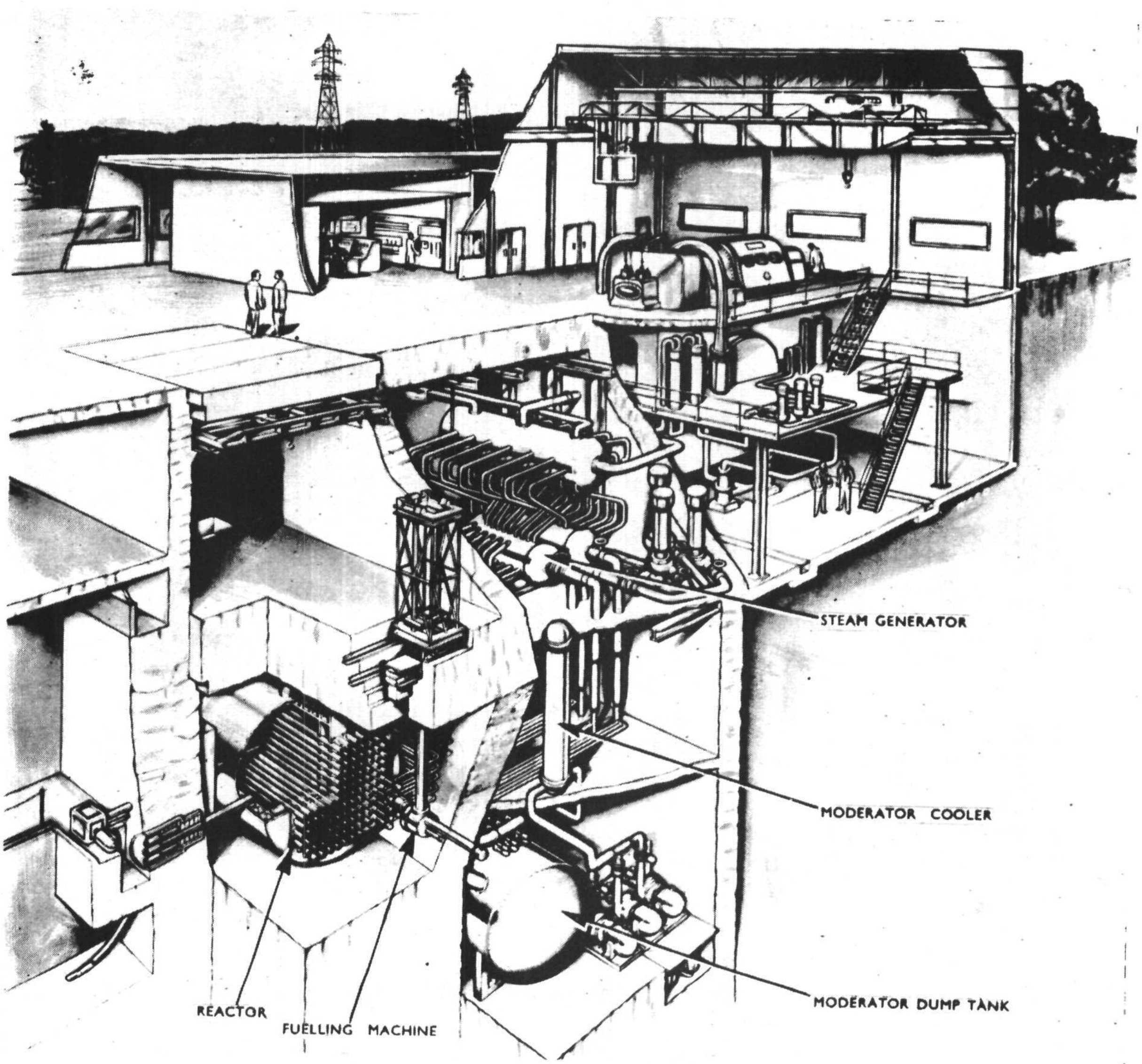


Fig. 7.10 — NPD-2 – general arrangement

Fuel element development
Zirconium alloy development
Pressure tube fabrication and performance
Initial and long term reactivity theory and experiments
Zircaloy-to-stainless steel joint development
Reactor system component testing
On-power refueling machine development
Demonstration reactor (NPD-2) construction.

Schedule

The NPD-2 is scheduled to go critical in the first half of 1961.

Economics

Power cost data for a prototype reactor plant are not significant. Also, since Canadian financing is on a different basis from that of the U.S., Canadian cost would be misleading if compared to U.S. numbers.

7.7 CANADIAN DEUTERIUM-URANIUM REACTOR (CANDU)

Atomic Energy of Canada Limited (AECL)

CANDU is a 200 MW_e nuclear station to be built at Douglas Point, Ontario and operated by Ontario Hydro. It formed the basis of the design of NPD-2 which is described in Section 7.6.

Objectives

CANDU is intended as a base-load station in the Ontario Hydro system. Economic studies indicate that it will produce competitive electrical power in the high fuel cost areas of Canada.

Concept Description

The reactor is heavy water-moderated and cooled. The fuel is natural uranium oxide in Zr-2 cladding tubes. The core tank is a double-walled aluminum cylinder with its axis horizontal. Thin-walled aluminum tubes run between the heads to provide 252 fuel positions. The moderator operates at about 200°F and is unpressurized. Reactor design characteristics are given in Table 7.1.

Reactor control has not been selected but will probably be by either moderator level or vertical absorber rods plus on-power refueling. On-power refueling will be used, providing high fuel burnup and reducing the excess reactivity under normal operating conditions.

Research and Development

The program mentioned in Section 7.6 for the NPD-2 is applicable to the CANDU reactor.

Schedule

The Douglas Point Station will operate in 1964 or 1965.

Economics

The power output of 200 MW_e was chosen on an economic basis. Preliminary studies, which had a target power cost of 5 mills/kwh, indicated that a 100 MW_e station could not be successful while a 150 MW_e plant would be marginal, success depending on whether a low-cost alloy could be used for fuel cladding. The 200 MW_e size apparently could meet the target with Zircaloy-2 fuel cladding. Larger ratings, although more economical, required a higher capital investment than considered justifiable for a first plant.

Since the Canadian design philosophy and economic structure differ from those of the United States, it is of interest to compare the results of cost studies conducted in the two countries.

Table 7.10 summarizes the economic factors that govern the estimates of reactor costs in the U.S. and Canada. If the U.S. cost estimates for a 200 MW_e, oxide-fueled, boiling reactor are converted to the Canadian basis, the following changes in power costs would result:[16]

1. capital investment decreases by 16%,
2. operating and maintenance costs decrease by 29%,
3. the power cost decreases by 49%.

A comparison of the estimated costs for constructing and operating the boiling reactor in Canada and CANDU are summarized in Table 7.11. Table 7.12 presents more detailed cost data for the S&L-NDA design on U.S. and Canadian bases plus CANDU. In addition to the economic ground rules given above, a number of adjustments were made in the plant capital costs to account for construction of the plant in Canada. These adjustments incorporate the following factors:

1. Based on a weighted average of labor rates in Toronto and Montreal, U.S. labor costs average 40% higher than those of Canada.
2. Since equipment sources for the Canadian plants are European and U.S. as well as Canadian, and the U.S. plants were estimated on the basis of U.S. equipment only, it was estimated that equipment costs average 10% higher in the U.S. than in Canada.
3. The dollar exchange rate was taken as a 3% U.S. penalty.
4. The computation of U.S. fuel costs on a Canadian basis assumes that the total fuel costs in Canada are similar to those for CANDU, as given in Table 7.8.
5. Operating, maintenance, labor, and materials costs were adjusted in accordance with the assumptions made in factors 1 and 2, above.
6. Insurance costs for the U.S. plant constructed in Canada were assumed the same as those for CANDU.

The Canadian system of accounting and the above capital cost adjustments have the following effects on the cost estimates for the 200 MW_e, boiling D_2O plant, if constructed in Canada.

1. The total plant investment decreases by approximately 16%.
2. The capital investment component of the generation cost decreases by about 59%.
3. The total operating and maintenance costs decrease by about 29%.

The total power cost of the 200 MW_e boiling reactor of U.S. design is estimated to be approximately 5.8 mills/kwh for plant construction and operation in Canada, as opposed to 11.3 mills/kwh* for the same plant constructed in the U.S. This value can be compared with the total power cost of 5.6 mills/kwh quoted for the CANDU by AECL.

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*

This figure is an earlier estimate made on U.S. basis and does not take into account the improvements that have allowed a reduction to 9.8 mills/kwh as reported elsewhere in this report.

Table 7.10 — Comparison of Bases for Computing Power Costs in the U.S. and Canada

	United States (S&L-NDA)	Canada (AECL-CANDU)
Plant Load Factor	0.8	0.8
Fixed Charges		
Reactor and auxiliaries, %/yr	14.0	9.31
Buildings and plant services, %/yr	14.0	6.13
Heavy water, %/yr	12.5	5.43
Fuel inventory charge		
Fabrication cost and non-nuclear material, %/yr	12.0	4.0
Fissionable material, %/yr	4.0	4.0
Heavy Water Cost, \$/lb	28.00	28.00
Fuel Element Costs, \$/kg of U (Both for Natural UO_2)		
Uranium cost	40.50	30.10
Pelletizing	12.50	8.80
Zr-2 and fabrication	48.75	30.10
Losses and shipping	7.80	—
Total Fuel Replacement Cost, \$/kg U	109.55	69.00

Table 7.11 — Summary of Estimated Costs of 200 MW_e Plants Built in Canada

	Estimated Costs for a 200 MW_e Boiling D_2O Plant of U.S. Design Built and Operated in Canada, mills/kwh	Estimated Costs for the 200 MW_e CANDU, mills/kwh
Plant Capital Cost	2.5	2.6
Heavy Water Inventory	0.7	0.5
Fuel Cost	1.6	1.1
Operation, Maintenance, and Supplies	1.0	1.4
Total Power Cost	5.8	5.6

Table 7.12 — Comparison of U.S. and Canadian Costs for 200 MW_e Plants

Design	S&L-NDA (U.S. Basis)			S&L-NDA (Canada Construction on Canada Base)			CANDU*		
Net Generation, kw	206,600			206,600			200,000		
Annual Generation at 0.8 L.F., kwh	1446×10^6			1446×10^6			1400×10^6		
	Investment, $	Annual Cost, $/yr	Power Cost, mills/kwh	Investment, $	Annual Cost, $/yr	Power Cost, mills/kwh	Investment, $	Annual Cost, $/yr	Power Cost, mills/kwh
Investment									
Equipment, materials and labor	44,181,000	6,185,000	4.277	36,439,000	2,608,000	1.81	36,813,110	2,700,000	1.93
Contingency (at 10%)	4,418,000	619,000	0.428						
	48,599,000	6,804,000	4.705	5,620,000†	402,000†	0.28†	5,683,590†	416,000†	0.297†
Escalation (at 12%)	5,832,000	816,000	0.565						
	54,431,000	7,620,000	5.270	42,059,000	3,010,000	2.09	42,496,700	3,116,000	2.227
Top charges (at 15%)	8,164,000	1,143,000	0.790	7,610,000‡	545,000‡	0.38‡	7,698,500‡	563,000‡	0.402‡
Total engineering and construction cost	62,595,000	8,763,000	6.060	49,669,000	3,555,000	2.47	50,195,200	3,679,000	2.629
D_2O	17,843,000	2,230,000	1.540	17,843,000	968,000	0.670	14,000,000	760,000	0.542
Total capital cost	80,438,000	10,993,000	7.600	67,512,000	4,523,000	3.140	64,195,200	4,439,000	3.171
Operation and Maintenance									
Fuel costs									
Inventory		134,000	0.092		226,000§	0.156§		138,000§	0.099§
Non-nuclear inventory		636,000	0.439						
Replacement		2,556,000	1.769		2,120,000	1.465		1,430,000	1.02
Total fuel cost		3,326,000	2.300		2,346,000	1.621		1,568,000	1.119
Heavy water									
Losses		358,000	0.247		358,000	0.247		195,000	0.139
Distillation plant operation		4,000	0.003		4,000	0.003		5,000	0.004
Total D_2O operating cost		362,000	0.250		362,000	0.250		200,000	0.143
Operating payroll		610,000	0.422		435,000	0.301		582,000	0.415
Maintenance – labor and materials		406,000	0.281		333,000	0.230		800,000	0.571
Supplies		140,000	0.097		127,000	0.088		98,000	0.070
Insurance		490,000	0.339		200,000	0.138		200,000	0.143
Total operation and maintenance cost		5,334,000	3.689		3,803,000	2.628		3,448,000	2.461
Total Capital and Operating Cost		16,327,000	11.289		8,326,000	5.768		7,887,000	5.632

* Capital and fuel costs received through personal contact with Chalk River on February 25, 1959. Other operating and maintenance costs were taken from AECL-557 (Jan. 1958).
† Contingency plus escalation at 15.4%.
‡ Top charges at 18.1%.
§ Total inventory.

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