MASTER

CIVILIAN POWER REACTOR PROGRAM

960

PART III

TID-8518(4)
Book 4

Status Report on Heavy Water Moderated Reactors as of 1959

UNITED STATES ATOMIC ENERGY COMMISSION

DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government nor any agency Thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

DISCLAIMER

Portions of this document may be illegible in electronic image products. Images are produced from the best available original document.

LEGAL NOTICE

This report was prepared as an account of Government sponsored work. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

TID-8518(4)
Book 4

CIVILIAN POWER REACTOR PROGRAM

PART III

*Status Report on Heavy Water Moderated Reactors
as of 1959*

Published 1960

UNITED STATES ATOMIC ENERGY COMMISSION

For sale by the Superintendent of Documents, U.S. Government Printing Office
Washington 25, D.C. - Price \$2.25

TABLE OF CONTENTS

	Page
SUMMARY.....	1
DISCUSSION.....	2
A. Objective.....	2
B. Concepts.....	2
C. Status of Reactor Construction and Operation.....	2
D. Technological Needs.....	4
1. Reactivity and Physics.....	4
2. Fuels and Materials.....	4
3. Heat Transfer.....	5
4. Coolant Stability and Temperature.....	5
5. Components.....	5
Appendix 1. STATUS OF DESIGN STUDY OF D ₂ O-MODERATED POWER REACTOR PLANTS.....	7
A. Concept Description.....	7
B. Objectives.....	8
C. Work Completed.....	8
1. General Research and Development.....	9
2. Experimental Reactors.....	9
3. Power Demonstration Reactors.....	9
D. Work Underway.....	11
E. Construction and Operation Schedule.....	11
F. Inherent Problems and Limitations.....	12
G. Potential Areas for Improvement.....	13
Appendix 2. STATUS OF STUDY OF D ₂ O-MODERATED POWER REACTORS.....	14
A. Concept Description.....	14
B. Objectives.....	14
C. Work Completed.....	14
1. General Research and Development.....	14
2. Experimental Reactors.....	16
3. Power Demonstration Reactors.....	17
D. Work Underway.....	18
1. General Research and Development and Experimental Reactors.....	18
2. Power Demonstration Reactors.....	19
E. Construction and Operation Schedule.....	19
F. Inherent Problems and Limitations.....	19
1. Physics.....	19
2. Fuel.....	19
3. Heat Transfer.....	19
4. Coolant Stability and Temperature.....	19
5. Components.....	19

Appendix 2—Continued	Page
G. Potential Areas for Improvement.....	20
Appendix 3. STATUS OF PROGRAM ON D ₂ O-COOLED-AND-MODERATED POWER REACTORS, NUCLEAR POWER DEMONSTRATION REACTOR (NPD), AND CANADIAN DEUTERIUM URANIUM REACTOR (CANDU)...	21
A. Concept Description.....	21
B. Objective.....	21
C. Work Completed.....	21
1. General Research and Development.....	21
2. Experimental and Power Demonstration Reactors.....	22
D. Work Underway.....	22
1. General Research and Development.....	22
2. Power Demonstration Reactors.....	23
E. Construction and Operation Schedule.....	23
F. Inherent Problems and Limitations.....	24
1. Reactivity and Physics.....	24
2. Fuel and Materials.....	24
3. Heat Transfer.....	24
4. Coolant Stability and Temperature.....	24
5. Components.....	24
G. Potential Areas of Improvement.....	24
Appendix 4. STATUS OF PROGRAM ON D ₂ O-COOLED-AND-MODERATED POWER REACTOR.....	27
A. Concept Description.....	27
B. Objectives.....	27
C. Work Completed and Underway.....	27
1. General Research and Development.....	27
2. Power Demonstration Reactors.....	28
D. Work Underway.....	30
E. Construction and Operation Schedule.....	30
F. Inherent Problems and Limitations.....	30
1. Reactivity and Physics.....	30
2. Fuel and Materials.....	30
3. Heat Transfer.....	31
4. Coolant Stability and Temperature.....	31
5. Components.....	31
G. Potential Areas of Improvement.....	31
1. Core Structural Materials.....	31
2. Critical Experiments.....	32
Appendix 5. STATUS OF PROGRAM ON THE SODIUM DEUTERIUM REACTOR (SDR).....	33
A. Concept.....	33
B. Objectives.....	33
C. Work Completed.....	33
1. General Research and Development.....	33
Appendix 6. REFERENCES.....	39

Tables

	Page
I. Characteristics of Heavy Water-Moderated Power Reactors	
	Face page 4
II. List of Irradiation Tests	20

Figures

1.1	Concept No. 3A, boiling D ₂ O, pressure tube, direct cycle ..	Face page 12
1.2	Performance diagram, concept No. 3A, boiling D ₂ O, pressure tube, direct cycle	Face page 12
1.3	Reactor arrangement, prototype Title I design	Face page 12
1.4	Composite flow diagram, prototype Title I design	Face page 12
2.1	Pressure vessel reactor. Liquid D ₂ O cooled; hot moderator (400 eMW)	Face page 20
2.2	Pressure tube reactor. Liquid D ₂ O cooled; cold moderator (460 eMW)	Face page 20
2.3	Pressure tube reactor. Boiling D ₂ O cooled; cold moderator (430 eMW)	Face page 20
2.4	Arrangement of HWCTR reactor vessel	Face page 20
2.5	Cooling system flow diagram for the HWCTR	Face page 20
3.1	Sketch of Nuclear Power Demonstration (NPD) reactor	25
3.2	Flow diagram of Nuclear Power Demonstration (NPD) reactor	26
4.1	Plant arrangement of Carolinas-Virginia Tube Reactor (CVTR)	Face page 32
4.2	Primary coolant system flow diagram for Carolinas-Virginia Tube Reactor (CVTR)	Face page 32
4.3	Tentative CVTR construction and operation schedule	Face page 32
5.1	Elevation layout of a single-region 200 eMW sodium-deuterium reactor	Face page 36
5.2	Schematic of a single-region 200 eMW sodium-deuterium reactor ..	37
5.3	Construction and operation schedule for the SDR program	38

SUMMARY

1. Estimates by both du Pont and Sargent and Lundy indicate that within the D_2O -moderated natural uranium-fuel reactor category, the lowest cost reactor will be cooled by boiling D_2O . The margin over nonboiling is small especially for reactors in the 100 eMW size and smaller. Atomic Energy of Canada, Ltd., on the other hand, is aiming their development effort at nonboiling D_2O as the coolant.

2. A major area in which research must be advanced before any of the reactor concepts can

be successful is fuel element fabrication and irradiation behavior. Both oxide and metallic elements have their own advantages and each type should be developed. Other areas in which additional work must be done are the reactivities of various fuel lattices, fuel element burnout especially for the boiling regime, zirconium pressure tubes and their connections to steel and methods of reducing and detecting D_2O leakage.

DISCUSSION

A. OBJECTIVE

The general purpose of all the programs reviewed in this report is to develop a reactor concept, utilizing natural uranium fuel and

heavy water moderator, that could lead to low-cost nuclear power. The specific objectives of each project are stated in somewhat more detail in the appendices.

<i>Concept</i>	<i>Designer</i>	<i>Type</i>	<i>No. of designs</i>	<i>Capacity eMW</i>	<i>Appendix</i>
Boiling D ₂ O-----	S&L-NDA----	pressure tube-----	2	69; 207	1
	du Pont-----	do-----	4	100-430	2
Liquid D ₂ O-----	do-----	pressure vessel-----	4	100-400	2
	do-----	pressure tube-----	4	100-460	2
	AECL-----	do-----	2	19; 200	3
	CVNPA-----	do-----	2	17; 270	4
Sodium Cooled-----	NDA-----	tube-----	2	10; 200	5

B. CONCEPTS

Listed in the table above are the reactor concepts whose status are summarized in this report. All of the concepts include thermal reactors that are moderated by heavy water and that, in plants of large electrical output, presumably can operate with natural uranium fuel.

A brief description of each reactor design is given in Part A of Table I. The status of each reactor design is given in detail in the appropriate appendix, listed above.

In all, 21 reactor designs are listed. In addition to differences in cooling mechanism, reactor construction, and plant capacity, major differences also exist in the choice of reactor fuel and in the powerplant. The du Pont reactors and one of the NDA reactors are fueled with uranium metal, whereas all the others are fuel with uranium oxide. Except for the S&L-NDA and the du Pont boiling-D₂O reactors, which generate steam in the reactor and pass it directly to the turbines, all the reactors have indirect steam cycles.

C. STATUS OF REACTOR CONSTRUCTION AND OPERATION

To date, construction has begun on only one of the D₂O-moderated power reactors, the AECL Nuclear Power Demonstration, NPD-2. No experience has been gained as yet in the operation of any D₂O-moderated power reactor. However, there is considerable experience available on the operation of D₂O-moderated production reactors. The Canadian NRU and NRX reactors and the U.S. Savannah River reactors fall in this latter category. A short history of the experience with heavy-water-moderated reactors follows.

The excellent properties of heavy water as a moderator for the fast neutrons of an atomic reactor were recognized very soon after the fission process was discovered. On the chance that graphite might not be a sufficiently good moderator to support a chain reaction, small-scale D₂O plants were built in 1943 so that a moderate supply of this material would be available. The plant at Trail, British Columbia, used the D₂-H₂O exchange process and the

three United States plants employed water distillation. The success of graphite as a moderator and the high production costs of D_2O (\$200/lb.) directed attention from this material almost as soon as the plants were completed. Canada, however, continued to place primary emphasis on D_2O -moderated reactors and designed the NRX reactor to use this material as the moderator.

The operating experience gained by Canada with their NRX reactor encouraged the United States to use D_2O as the moderator for the five large production reactors that were built at Savannah River beginning in 1951. The very large amount of heavy water required for the production reactors was obtained by building a plant at Dana, Indiana, and a plant at Savannah River, each capable of producing 500 tons/year of D_2O . The H_2S - H_2O dual-temperature-exchange process was employed and proved to be so superior to the processes previously used that the cost of producing D_2O was reduced to the point that D_2O is now sold for \$28/lb.

Designers in the power reactor field recognized that D_2O moderation would allow sufficient excess multiplication that the reactors could be fueled with natural uranium. This latter feature is of major importance to those countries that do not have access to plants for the enriching of the U-235 isotope of normal uranium.

Canada is now building a 20 eMW power reactor that will be moderated with heavy water and has authorized construction of a 200 eMW reactor of this type, which is currently being designed. Several pilot and test reactors of a similar type are under construction, both in this country and abroad including the Heavy Water Components Test Reactor at Savannah River.

Much new technology needs to be developed, however, before these reactor types can produce low-cost electric power. The chief problem is related to the higher temperatures required for the production of electricity. The Savannah River production reactors have the moderator

tank at atmospheric pressure and therefore can heat the water that discharges into this space to only about 100°C. Power reactors require a temperature of at least 200°C and preferably, considerably higher. If the reactors are cooled with D_2O , this high temperature requirement forces the designers to employ pressures of 500–1,500 lbs./sq. in.—a not too difficult item in itself, but when coupled with the fact that zirconium is almost the only material of construction that can be used for the pressure tubes entering the reactors, the problems are increased several-fold.

Another major difference is in the cladding of the fuel. All production reactors in the Western group of nations currently use either aluminum or magnesium for this service. Neither of these materials is very corrosion resistant in 200°C water. Zirconium is being used as the cladding in most of the power reactors that are cooled with either D_2O or H_2O . Again new technology must be and is being developed.

The production reactors, however, have given much operating experience that should be helpful to the designers of a D_2O -moderated power reactor. For example, these reactors circulate large quantities of D_2O at moderate temperatures and pressures through heat exchangers, valves, pumps, etc., and the annual loss of D_2O has been only about 3 percent per year, based on the total D_2O in the system.

The D_2O isotopic purity of the moderator at Savannah River has been maintained at above 99.7 percent via a distillation procedure. The leakage rate of H_2O has been in the range of 6 lbs./day requiring 6,000 lbs./hr. of low pressure steam for its removal. Because the pressure in a D_2O -moderated power reactor will be higher than in a production reactor, the H_2O leakage would be expected to be smaller. On the other hand, the higher D_2O pressure will probably affect D_2O losses adversely. Other moderator problems, including radiolytic decomposition and entry of ionic contaminants, have been easily kept under control. Tritium in the moderator presents a minor health haz-

ard when this material is spilled, but the usual protective devices, such as gloves and aprons, are generally sufficient to allow safe working in the area.

In addition to this experience with heavy water moderator, some experience has been obtained with each of the cooling mechanisms that are being suggested for the D_2O -moderated power reactors. Several liquid-water-cooled and boiling-water-cooled reactors have been built and operated successfully in the United States and Canada. Among the water-cooled reactors are the Hanford and Savannah River production reactors, Shippingport Atomic Power Station (PWR), Materials Testing Reactor (MTR), and the Canadian National Research Universal (NRU), and National Research Experiment (NRX); among the boiling-water reactors are the Experimental Boiling Water Reactor (EBWR) and the Vallecitos Boiling Water Reactor (VBWR). Experience in the United States with gas cooling has been gained with the Oak Ridge X-10 and Brookhaven research reactors. Experience with sodium cooling has been gained in the operation of the graphite-moderated Sodium Reactor Experiment (SRE). The greatest amount of experience, however, has been with liquid-water cooling.

Thus, much of the experience with heavy-water-moderated production reactors is applicable to power reactors moderated with the same material. However, there are still sufficient differences between the production and power reactors, especially in the fuel elements and materials of construction, to require a significant amount of research and development effort.

D. TECHNOLOGICAL NEEDS

The particular research and development items that are required for each reactor are given in the appendices. The purpose of this section is to outline briefly the major general technical unknowns on which research and development is required for the D_2O -moderated power reactors. Some of these items were men-

tioned in the discussion of the experience with heavy-water-moderated reactors.

1. Reactivity and Physics

The nuclear properties of a wide variety of cold-clean natural uranium lattices in heavy water moderator have been measured. However, most of the data were for metal-fuel lattices. In addition, the temperature of the moderator was generally about room temperature. The nuclear design of almost all the reactor concepts presented in this report was extrapolated, by a variety of computational techniques, from these experimental data.

Additional physics studies, along the following lines, are therefore needed:

(a) The nuclear properties of oxide-fueled lattices should be more thoroughly surveyed.

(b) The properties of lattices with hot D_2O should be measured.

(c) To obtain reliable measures of the properties of the more promising power reactors, the properties of fairly accurate mockups of the reactor lattices should be experimentally determined.

(d) The transient response of each power reactor to sudden change in temperature, reactivity, or load should be carefully analyzed.

2. Fuels and Materials

Although extensive testing of prototype fuel elements has been under way and much testing of Zircaloy has been done, the following list summarizes the major areas in which considerable research and development effort may be fruitful.

(a) Irradiation testing of prototype metallic fuel elements is necessary in order to prove their adequacy to withstand long periods of exposure to neutrons under conditions comparable to those expected in the power reactor designs.

(b) A lesser program of irradiation testing is required for the oxide than for the metal because of the generally better irradi-

TABLE I
D₂O MODERATED POWER REACTORS - DESCRIPTION AND DESIGN INFORMATION

CLASSIFICATION			Boiling D ₂ O					Pressurized Liquid D ₂ O												Sodium Cooled			
DESIGNER	Sargent & Lundy - NDA		du Pont					du Pont								AECL		Carolinas-Virginia		NDA			
DESIGN CODE			1K	1K-200	1K-300	1K-430	1B		1B-200	1B-300	1B-400	1D	1D-200	1D-300	1D-460	NPD-2	CANDU	CVTR	CVNPA	CHUGACH	SDR(3)		
A. DESCRIPTION																							
Category	natural U	natural U	natural U	natural U	natural U	natural U	natural U		natural U	natural U	natural U	natural U	natural U	natural U	natural U	natural U	natural U	enriched U	natural U	enriched U	natural U		
Reactor Design	pressure tube	pressure tube	pressure tube	pressure tube	pressure tube	pressure tube	pressure vessel		pressure vessel	pressure vessel	pressure vessel	pressure tube	pressure tube	pressure tube	pressure tube	pressure tube	pressure tube	pressure tube	pressure tube	tube	tube		
Moderator	cold	cold	cold	cold	cold	cold	hot		hot	hot	hot	cold	cold	cold	cold	cold	cold	cold	cold	cold	cold		
Coolant Circulation	forced	forced	forced	forced	forced	forced	forced		forced	forced	forced	forced	forced	forced	forced	forced	forced	forced	forced	forced	forced		
Fuel Material	UO ₂	UO ₂	U metal	U metal	U metal	U metal	U metal		U metal	U metal	U metal	U metal	U metal	U metal	U metal	UO ₂	UO ₂	UO ₂	UO ₂	UO ₂	metallic U		
Fuel Shape	rods	rods	tubes	tubes	tubes or rods	tubes or rods	tubes		tubes	tubes or rods	tubes or rods	tubes	tubes	tubes or rods	tubes or rods	rods	rods	rods	rods	rods	rods		
Fuel Cladding	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy		Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy	stainless steel	Zircaloy		
Fuel Cycle, % core change per loading	20%	25%	100%	100%	100%	100%	100%		100%	100%	100%	100%	100%	100%	100%	≤ 100%	≤ 100%	100%	100%	100%	~ 10%		
Neutron Energy Spectrum	slow	slow	slow	slow	slow	slow	slow		slow	slow	slow	slow	slow	slow	slow	slow	slow	slow	slow	slow	slow		
Plant Steam Cycle	direct	direct	direct	direct	direct	direct	indirect		indirect	indirect	indirect	indirect	indirect	indirect	indirect	indirect	indirect	indirect	indirect	indirect	indirect		
Steam Condition	saturated	saturated	saturated	saturated	saturated	saturated	saturated		saturated	saturated	saturated	saturated	saturated	saturated	saturated	saturated	saturated	superheated*	superheated*	superheated	superheated		
Site	SRP	SRP	SRP	SRP	SRP	SRP	SRP		SRP	SRP	SRP	SRP	SRP	SRP	SRP	Ontario, Can.	Ontario, Can.	Parr, S. C.	Southeast USA	Alaska	-		
Plans & Flow Diagrams	-	Figs. 2.1 & 2.2	-	-	-	Fig. 3.3	-		-	-	Fig. 3.1	-	-	-	Fig. 3.2	Figs. 4.1 & 4.2	-	Figs. 5.1 & 5.2	-	-	Figs. 6.1 & 6.2		
B. OBJECTIVE	study	study	study	study	study	study	study		study	study	study	study	study	study	study	power demonstr.	study	study	study	study	study		
C. CONSTRUCTION SCHEDULE	none	none	none	none	none	none	none		none	none	none	none	none	none	none	1961 (App. 4)	none	none	none	none	none		
D. OPERATING EXPERIENCE	none	none	none	none	none	none	none		none	none	none	none	none	none	none	none	none	none	none	none	none		
E. RESEARCH AND DEVELOPMENT	Appendix 2	Appendix 2	Appendix 3	Appendix 3	Appendix 3	Appendix 3	Appendix 3		Appendix 3	Appendix 3	Appendix 3	Appendix 3	Appendix 3	Appendix 3	Appendix 3	Appendix 4	Appendix 4	Appendix 5	Appendix 5	Appendix 6	Appendix 6		
F. DESIGN PARAMETERS																							
Electric Megawatts - Net	69.1	206.6	100	200	300	430	100		200	300	400	100	200	300	460	19.3	200	17.0*	270	12.6	200		
Electric Megawatts - Gross	73.6	220	106	212	318	456	106		212	318	424	106	212	318	487	22.0	203.4	18.9*		13.4			
Thermal Megawatts	255	790	520	810	1220	1740	440		855	1280	1720	435	980	1440	2220	88.9	794	61.8		43.	678		
Annual Electric Generation, net KWH/yr	484 x 10 ⁶	1448 x 10 ⁶	701 x 10 ⁶	1402 x 10 ⁶	2103 x 10 ⁶	3013 x 10 ⁶	701 x 10 ⁶		1402 x 10 ⁶	2103 x 10 ⁶	2804 x 10 ⁶	701 x 10 ⁶	1402 x 10 ⁶	2103 x 10 ⁶	3224 x 10 ⁶	135 x 10 ⁶	1402 x 10 ⁶	119 x 10 ⁶		88.3 x 10 ⁶	1402 x 10 ⁶		
Over-all Plant Efficiency, %	27.1%	26.2%	19.2%	24.7%	24.6%	24.7%	22.7%		23.4%	23.4%	23.3%	23.0%	20.4%	20.8%	20.7%	21.7%	25.2%	-		29.3%	29.5%		
Reactor Operating Pressure, psig	805	825	250	815	815	815	524		783	783	705	800	783	783	823	1069	~ 1000	1405	61	~ 60			
Coolant Outlet Temperature, °F	516°	515°	397°	518°	518°	518°	451°		482°	482°	477°	491°	482°	482°	486°	530°	525°	555°		950°	800°		
Coolant Average Temperature Rise, °F	17°	17°	27°	59.4°	59.4°	59.4°	48.6°		68.4°	68.4°	63.0°	45.0°	68.4°	68.4°	72°	45°	95°	50°		300°	200°		
Core Size - diameter, ft	12.0	16.4	11.8	13.1	15.5	19.2	10.5		12.0	13.0	14.0	10.8	11.4	13.8	15.8	11.0	18.0	6.2		7.5	16.1		
Core Size - height, ft	11.1	17.7	15	15	15	15	15		15	15	15	15	15.0	15.0	15.0	13.0	16.6	8		9.4	20		
Core Inventory - metric tons fuel	19.0	67.7	24.8	41.0	58	81	26.5		27.2	32.0	35.5	17.6	29.1	43.6	56.3	17	62	4.58		2.01	117		
Core Inventory - metric tons U	16.8	59.7	24.8	41.0	58	81	26.5		27.2	32.0	35.5	17.6	29.1	43.6	56.3	15	54.6	4.04		1.77	117		
Initial Fuel Enrichment - wt % U-235	0.71%	0.71%	0.71%	0.71%	0.71%	0.71%	0.71%		0.71%	0.71%	0.71%	0.71%	0.71%	0.71%	0.71%	0.71%	0.71%	2.0%		3.0%	0.71%		
Spent Fuel Enrichment - wt % U-235	0.29%	0.30%	0.48%	0.46%	0.46%	0.45%	0.48%		0.49%	0.49%	0.50%	0.46%	0.48%	0.47%	0.46%	?	?	1.57%		2.0%	~ 0.3%		
Design Exposure - average MWD/metric ton U	6100	7500	3580	4180	4400	5500	3580		3960	3960	3850	3850	3520	3850	4180	5400	8100	3150		7500	4000		
Average Conversion Ratio - gm Pu/kg U	2.8	2.5	2.4	2.8	2.9	3.1	2.5		2.6	2.6	2.6	2.4	2.5	2.7	2.8	3 to 4	~ 3	1.72		?	3.01		
Average Specific Power - thermal KW/metric ton of fuel	13,400	11,700	21,000	19,800	21,000	21,500	16,600		31,400	40,000	48,500	24,700	33,700	33,000	39,400	5,220	12,800	13,500		21,400	5800		
Average Power Density - thermal KW/ft ³ of core	203	211	317	400	430	400	338		503	642	745	316	640	641	754	71.8	183	255		104	166		
Peak to Average Power Ratio	2.13	3.00	2.22	2.22	2.00	2.00	2.32		2.32	2.32	2.32	1.85	2.32	2.32	2.00	4.24	3.19	2.90		2.32	3.78		
Central Fuel Temperature (max.) °F	4000°	4000°					880°					990°				2900°	4000°	4700°		4160°	900°		
Average Heat Flux, Btu/hr/ft ²	113,000	98,000	150,000	162,000	180,000	180,000	324,000		335,000	341,000	380,000	467,000	341,000	341,000	395,000	73,000	109,000	118,000		138,000	61,400		
Maximum Heat Flux, Btu/hr/ft ²	240,000	294,000	333,000	360,000	360,000	360,000	752,000		777,000	791,000	881,000	863,000	791,000	791,000	791,000	158,000	330,000	343,000		321,000	232,000		
Steam Temperature, °F	510°	510°	397°	517°	517°	517°	367°		375°	375°	375°	410°	375°	375°	375°	450°	not established	725°			730°		
Steam Pressure, psia	765	765	240	795	795	795	167		185	185	185	277	185	185	185	415		415			1000		
Turbine Back Pressure, in. Hg	1.5	3.5	1.5	1.5	1.5	1.5	1.5		1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5		1.5			1.5		
Cooling Water Temperature, °F		95°	65°	65°	65°	65°	65°		65°	65°	65°	65°	65°	65°	65°								

* fossil-fueled superheater

ation behavior of the oxide. However, all oxide fuels, especially those manufactured by new techniques (e.g., explosive compaction or coextrusion) should be proof-tested.

(c) Considerable effort could be expended on developing improvements in quality and reductions in cost of the fabrication processes for both oxide and metal fuel elements.

(d) The irradiation testing of pressure tubes and their joints with other materials is highly essential for reliable design of all pressure tube reactors, including the mechanical and hydraulic testing of these items, out of pile, would also be of much worth. In addition, development of a low-cost high-quality fabrication process for the pressure tubes and their joints would assist significantly in reducing the cost of several promising reactors.

(e) Corrosion testing of fuel elements, pressure tubes, housing tubes, and other important components of the reactor should be continued in order to assure the safe operation of a given design.

3. Heat Transfer

All of the power reactor designs are probably designed with adequate margins of safety from heat transfer burnout. The major experimental work in this field, to date, has been with subcooled liquid coolant. In order to determine the potential, or the safe limits of operation, of various reactor fuel elements that are cooled by a boiling fluid, it will be necessary to expand the experimental program on boiling heat transfer burnout.

4. Coolant Stability and Temperature

No major effort is required on this subject for the power reactor concepts shown in this report.

5. Components

All unusual mechanisms, seals, drives, refueling machines, closures, etc., must be tested before a given reactor concept is completed in order to assure safe and reliable operation.

Appendix 1—STATUS OF DESIGN STUDY OF D₂O-MODERATED POWER REACTOR PLANTS

Sargent & Lundy, Engineers—Nuclear Development Corporation of America (S&L-NDA)

The United States Atomic Energy Commission awarded a contract to Sargent & Lundy of Chicago, Illinois, on November 6, 1958, to perform a design study of heavy-water-moderated power reactor plants. This study was completed on March 1, 1959 and supplemental studies were completed on June 15, 1959. Sargent & Lundy retained Nuclear Development Corporation of America of White Plains, New York, as subcontractor for the principal nuclear aspects of the design.

Comparisons and evaluations of various concepts of heavy-water-moderated power reactor plants led to the conclusion that the concept holding the greatest promise for producing low-cost nuclear power in the near future is one that is cooled by boiling heavy water with the steam generated being passed directly to a steam turbine-generator. In large capacity units the preferred reactor configuration is one in which pressure tubes contain the coolant fluid while the moderator is maintained in a lower pressure calandria. The study indicated a prototype of the recommended power plant could be built by 1963, using current technology plus that development which can be carried on in parallel with the design and construction of a prototype.

This appendix contains discussions of the designs of the 207 eMW plant and the 69 eMW prototype that resulted from the Sargent & Lundy study.

A. CONCEPT DESCRIPTION

See Table I of the body of this report.

A drawing of the S&L-NDA conceptual design of a boiling-D₂O-cooled, natural-uranium-

oxide-fuel, D₂O-moderated power reactor is presented in Figure 1.1. A flow diagram of the direct-cycle saturated-steam plant that is associated with the reactor is shown in Figure 1.2.

The full scale plant recommended as a result of the study of various D₂O-moderated reactor concepts utilizes a direct cycle, boiling D₂O reactor-turbine system to produce 207 eMW net output. D₂O steam is produced in the tubes of a pressure tube type reactor. The major features of the plant are summarized in the following paragraphs.

The reactor consists of a low-pressure aluminum calandria containing cold D₂O moderator. Thin wall aluminum calandria tubes are arranged in a regular hexagonal lattice and contain 268 vertical Zircaloy-2 pressure tubes. The pressure tubes pass through the calandria, inside aluminum calandria tubes, and contain the fuel elements.

The coolant enters the pressure tubes at the bottom of the reactor through a header-pigtail distribution system at 475° F. Its temperature increases to 515° F in passing through the tubes. At the top of the tubes, the steam quality is 14 percent. The steam-water mixture rises to drums above the reactor where the steam is separated and flows directly to the turbine at 765 p.s.i.a., in a dry and saturated condition. Approximately 16.0×10^6 lbs. per hours of D₂O are force-recirculated from the steam drums to the reactor inlet. Total steam flow to the turbine is 2.54×10^6 lbs./hr. for 207 MW net electrical output. After condensing, the D₂O coolant is returned to the reactor by means of condensate pumps and reactor feed-water pumps. One stage of feed-water heating

is provided to heat the D₂O feed water to 194° F before it is mixed with the recirculated water being returned to the reactor. The mixed mean temperature of the reactor feed-water is 475° F.

The pressure tubes and the calandria tubes are separated by an annular air gap which provides an insulating barrier between the high temperature coolant and the cold moderator. As a result, the moderator can be maintained at low temperature and pressure. Moderator cooling pumps and heat exchangers are provided to maintain a maximum moderator temperature of 200° F. Separation of the moderator and coolant also reduces the possibility of contamination of the moderator by H₂O or other impurities.

The fuel elements are 1/2-inch diameter UO₂ rods clad with Zircaloy and arranged in 37-rod clusters. They operate at a maximum central fuel temperature of 4,000° F, a maximum clad temperature of 550° F, and maximum heat flux of 325,000 B.t.u./hr.-ft.² All three values represent conservative design practice and are well within the realm of present day technology.

Reactivity is controlled by 12 shim rods and 7 regulating rods. Nineteen additional safety rods are provided for scram. The rods are actuated from the top of the reactor.

The reactor, steam drums, and auxiliaries are located in a 190-foot diameter spherical steel containment shell designed for a pressure of 25 p.s.i.a. Steam leaves the reactor building and flows to the adjoining turbine building. The turbine building houses the turbine-generator, condenser, feed pumps, and turbine auxiliaries as well as the offices and shops.

The turbine is a nominal 220 eMW, 1,800-r.p.m., tandem compound, double flow, extraction steam turbine designed for operation with saturated steam at 765 p.s.i.a. and exhausting at 3 1/2 inches of Hg absolute. Conventional materials are used throughout the turbine and condenser, except for the turbine seals.

The condenser is a single pass, double tube sheet design with aluminum tubes. Piping in

the main steam and condensate system is carbon steel. Corrosion products are removed by full flow condensate demineralizers. Glands on the valves, pumps, and turbine are of a special design to limit and collect any D₂O leakage.

B. OBJECTIVES

The study had two objectives: (1) selection and recommendation of a conceptual design of a large-scale heavy-water-moderated power reactor plant providing optimum economics with slight fuel enrichment, but capable of operating with natural uranium fuel; and (2) a preliminary design and cost estimate for a prototype power reactor plant of the minimum size necessary to demonstrate the recommended concept, including operability on natural uranium fuel. Each of these designs was to be based on currently available technology, and developments which could be undertaken concurrently with plant design and construction. Operation of the prototype was to be in 1963.

C. WORK COMPLETED

In this section of the appendix, the design of a full-scale, 200 eMW powerplant is discussed.

The project was divided into two parts in accordance with the twofold objective. Part I consisted of an evaluation of various reactor plant concepts, resulting in the selection of the most promising plant concept and size, and determination of the prototype design criteria. In Part II, preliminary design of the prototype plant was undertaken. This section contains the results of the studies conducted in Part I of the project.

Part I of the design study was conducted in the following phases:

(a) Qualitative analysis of conceivable heavy water-moderated reactor types led to the selection of six concepts which appeared to offer the greatest probability of optimum economics as well as construction of a prototype based on currently available technology, and/or technology which could be

developed in parallel with the prototype design and construction for operation in 1963.

(b) Conceptual design of the six selected plants, utilized as much of currently available information as possible, in sufficient detail to permit the preparation of reasonable construction and operating cost estimates.

(c) Evaluation and comparison of the six designs in terms of present and expected energy costs included the effects of using slightly enriched and natural uranium fuels, performance characteristics, component fabrication, and construction problems.

(d) The plant concept with the greatest potential for economic nuclear power within the terms of the design criteria established by the Atomic Energy Commission was selected. (See Section A, this appendix.)

(e) Determination of design criteria for the minimum size prototype power plant necessary to demonstrate the selected concept.

1. General Research and Development

No experimental or development work was done specifically for the S&L-NDA study. The scope of the study and the short time available for its preparation required that maximum use be made of existing information. To this end appropriate data already developed were used wherever possible. Information developed at Savannah River Operations Office (du Pont Co.), Atomic Energy of Canada, Ltd., Argonne National Laboratory and at Nuclear Development Corporation of America was particularly useful. In addition, the results of informative visits made to Hanford Atomic Products Operations, Chase Brass Co., and Atomics International were used.

2. Experimental Reactors

None.

3. Power Demonstration Reactors

The design of a 69 eMW prototype of the boiling-D₂O-cooled power reactor, of the type

described in Section A, this appendix, is discussed in this section.

a. Description

See Table I of the body of this report.

The prototype power reactor plant duplicates the general concept as well as the more important specific features of the 200 eMW plant recommended on the basis of the Part I program. In the prototype plant, a boiling D₂O, pressure tube, natural uranium oxide fueled reactor generates D₂O steam at 515° F and 795 p.s.i.a. which passes directly to a turbine to produce a net output of 69.1 eMW. The principal features of the prototype plant exterior and site are a cylindrical steel reactor containment building, separate turbine and fuel handling buildings adjacent to the reactor building, and the relatively isolated radioactive waste handling facility.

Reactor.—The reactor, located in the reactor containment building, consists of a low-pressure aluminum calandria which contains the cold D₂O moderator and which is pierced by vertical aluminum calandria tubes arranged in an equilateral triangular lattice array. Zircaloy-2 pressure tubes passing through the calandria tubes contain the boiling D₂O coolant and the fuel elements.

The coolant and moderator are operated in completely independent systems. The air-filled annulus between the pressure tube and calandria tube provides thermal insulation between the hot coolant and the cold moderator.

As in the 200 eMW plant, coolant D₂O enters a header system below the reactor, passes through individual connecting pipes to the pressure tubes, passes upward through the pressure tubes where it is heated by the fuel elements, and passes out through an upper header system to two steam drums with integral steam separators. Dry and saturated steam from the drums is delivered to the turbine, while liquid D₂O is force-recirculated to mix with the turbine cycle condensate return at the reactor inlet.

Reactor control is effected by shim and regulating rods driven from below the reactor; emergency shutdown is achieved by release of safety rods latched from above and, as a back-up if necessary, by fast drain of the moderator D_2O .

Refueling is accomplished by a remotely operated machine located in a shielded room above the reactor. With the reactor shut down and cooled to atmospheric pressure, the machine removes seal plugs on the ends of pressure tube extensions to gain access to the fuel elements.

Reactor Dimensions.—The local lattice cross section is chosen to be identical with that of the 200 eMW plant in order that maximum information on nuclear performance of the large reactor may be obtained from operation of the prototype. Accordingly, the following dimensions are chosen as common to the two plants:

Fuel element

Pellets (natural UO_2), OD, 0.5 inch.

Clad, OD, 0.552 inch.

Thickness, 0.025 inch.

No. rods in cluster, 37.

Pressure tube (Zircaloy-2), OD, 4.974 inches.

Thickness, 0.162 inch.

Calandria tube (Aluminum), OD, 5.875 inches.

Thickness, 0.060 inch.

Triangular lattice spacing, 11.1 inches.

Overall size of the prototype reactor was chosen to be a minimum consistent with reasonable operation on natural uranium fuel. To achieve an initial excess reactivity of 0.01 with equilibrium poison conditions at operating temperature, a calandria size of approximately 16-foot diameter by 16-foot height, containing about 81 metric tons of D_2O moderator and reflector, was selected, as compared with 140 tons in the 200 eMW calandria. In order that power output not be excessive at limiting operation of the fuel elements, the active core volume is minimized by use of a 2-foot thickness of the moderator as a reflector, resulting in a core containing 152 lattice positions of 11.1-foot active height. Of these, 133 lattice tubes contain fuel elements. Each fuel tube contains two fuel elements, each of 5.5-foot

length. With two elements per channel, axial inversion refueling schemes may be used as in the larger reactor. The 19 shim and regulating control rods occupy the remaining lattice positions, while the 19 safety rods occupy smaller diameter tubes located between lattice positions.

Reactor Operating Conditions.—Operating conditions in the prototype have been selected so as to duplicate to the greatest possible extent the operation of the 200 eMW reactor. Because all lattice dimensions and materials are identical in the two reactors, the important temperature and stress limitations remain unchanged. Therefore, maintaining the same coolant temperature and pressure conditions, as well as fuel element temperature limits, will lead to comparable operating experience both within the reactor and in the steam cycle. Accordingly, the following conditions have been specified for the prototype:

Coolant pressure at steam drum, 795 p.s.i.a.

Coolant temperature at steam drum, 515°F.

Coolant exit flow quality from core, 14%.

Feed-water temperature, 387°F.

Fuel element central temperature, 4,000°F.

These conditions lead to essentially identical core inlet temperatures and degrees of subcooling at the reactor inlet, so that thermodynamic and hydrodynamic conditions in the pressure tubes are in comparable ranges for the two reactors.

For the prototype these conditions lead to a thermal power of 225 MW, a steam flow rate of 1.06×10^6 lbs./hr. and a recirculation flow rate of 6.48×10^6 lbs./hr.

The moderator cooling system for the prototype is capable of maintaining the calandria D_2O at an average temperature of 120°F; this temperature is lower than that of the large reactor in order to increase the excess reactivity available in the prototype when operating on natural uranium.

Excess reactivity above that required to compensate for equilibrium poisons at operating temperature is 0.01 with natural uranium oxide fuel. The excess reactivity is increased

to 0.04 by use of fuel with slight enrichment (0.77 percent).

Turbine Plant.—The dry and saturated D₂O steam generated in the reactor is carried directly to a turbine-generator located in the turbine building where it is used to generate 73.6 eMW. As in the recommended 200 eMW plant, the steam, after passing through the turbine, is condensed and returned to the reactor by means of condensate pumps and feed-water pumps. Four stages of extraction steam feed-water heating are used to bring the final feed-water temperature to 387° F before it is mixed with the reactor recirculation water. Again the number of stages of feed-water heating and the final feed-water temperature correspond to those of the recommended 200 eMW plant.

Although the cycle and operating inlet and outlet temperatures of the D₂O are identical in the prototype and the 200 eMW plant, a difference in plant efficiencies exists because of the differing turbine characteristics. The prototype condenser operates at 1.5 inches of Hg absolute, resulting in a new plant efficiency of 27.0 percent while that of the 200 eMW plant operating at 1.5 inches of Hg absolute has an efficiency of 27.7 percent.

The prototype turbine is a nominal 70 eMW, 3,600-r.p.m., tandem compound, triple flow, extraction steam turbine designed for operation with saturated D₂O steam at 765 p.s.i.a. exhausting at 1½ inches of Hg absolute. Conventional materials are used throughout the turbine and condenser, except for the turbine seals.

The condenser is of two pass, double tube sheet design with aluminum tubes. Glands on the valves, pumps, and turbine are of special design to limit and collect any D₂O leakage.

Auxiliary Systems.—The auxiliary systems, consisting of the moderator cooling system, the purification systems, the shield cooling system, the service and circulating water system, waste disposal system, D₂O recovery system and the D₂O distillation plant for the prototype are all similar, except for size, to those proposed for the 200 eMW plant.

Buildings.—The reactor, steam drums, and reactor auxiliaries are housed in a 125-foot diameter by 175-foot high cylindrical steel reactor containment building. The turbine-generator, condenser, and condensate cycle auxiliaries are located in a separate turbine building adjacent to the reactor building. The offices, control room, shops, locker rooms, and laboratories are a part of the turbine building. In addition, a fuel handling building and waste disposal building are provided.

In general the prototype buildings are similar, except for size, to those of the 200 eMW plant. Because of the prototype size, very few changes in building requirements and number of buildings provided can be made when comparing the two plants.

b. Purpose

The purpose of the prototype plant is to demonstrate the operability of a boiling-D₂O-cooled, direct cycle powerplant that is fueled with natural uranium and moderated with heavy water.

c. Significant Design Parameters

See Table I of the body of this report and the description given above in 3.a.

d. Cross-Section View and Flow Diagram

See figures 1.3 and 1.4.

e. Operating Experience

None.

D. WORK UNDERWAY

At present, no work is in progress. The design study has been completed.

E. CONSTRUCTION AND OPERATION SCHEDULE

No authorization was received for construction of the full-scale 200 eMW reactor plant or the prototype plant.

F. INHERENT PROBLEMS AND LIMITATIONS

Construction of a prototype reactor would require a development program that could be carried on in parallel with the design and construction of the plant. The major development areas for the prototype, as they appeared to S&L-NDA, are outlined below.

PRIMARY DEVELOPMENT ITEMS

a. Pressure Tube Fabrication.—In the past a number of different size Zirconium and Zircaloy-2 tubes have been made with diameters from 1 in. to $4\frac{1}{2}$ in., thicknesses from 0.030 in. to 0.069 in., and lengths up to 57 ft. No tube produced to date has combined the 5 in. O.D., 0.162 in. wall, and 24 ft. length required for the prototype pressure tubes; however, tube length appears to present no problems. Tubes 57 ft. long, $3\frac{1}{4}$ in. diameter, and $\frac{1}{4}$ in. thick are being produced in quantity by Chase Brass Co., and Bridgeport Brass Co., in conjunction with Tube Reducing Co. While no tubes of 5 in. diameter and 0.162 in. thickness have yet been produced in quantity, Tube Reducing Co., Harvey Aluminum Co., Curtiss-Wright Co., Chase Brass Co. and Bridgeport Brass Co., all have stated emphatically that such tubes can be made. Development of tools will be required to produce satisfactory tubes, but past experience indicates that this should be straightforward.

b. Joints for Zircaloy-2-to-Stainless-Steel Tube Connections.—It is necessary to join the Zircaloy-2 pressure tubes to the stainless steel coolant distribution system above and below the core. Metallurgically bonded joints have been investigated in the past and do not appear to be sufficiently developed for consideration at this time. A number of mechanical joints are available today which do appear to have suitable characteristics for this use. One, which is being tested at Hanford for Plutonium Recycle Test Reactor is the Marman "Conoseal." Other designs must also be investigated and evaluated. There does not seem to be any evi-

dent reason why a suitable mechanical joint should not be attainable.

c. Turbine Seals.—In the direct cycle unit it is important that unrecoverable loss of D_2O at the turbine seals be minimized. This will require the development of suitable shaft seals for the turbine. In this seal system it is also important that in-leakage of H_2O be held to a low rate.

d. Critical and Exponential Experiments.—Whether extensive critical or exponential experiments, beyond those already available, will be required for this program is not clear at this time. A decision on the need for such experiments cannot be made until further prototype design is performed.

SECONDARY DEVELOPMENT ITEMS

a. Coolant Tube End Closures.—End closures of suitable leak tightness must be developed for the coolant tubes. These end closures will have to be suitable for repeated opening and closing for refueling and for thermal cycling over long periods of time. Such closures are presently under development at a number of sites with promising results.

b. Calandria.—The calandria is a large unpressurized vessel with numerous lattice positions mounted within it. Similar calandria are used in NRX and will be used in PRTR and NPD-2. Suitable manufacturing and assembly techniques for the calandria may require some minor developments. The joining of calandria tubes to the end plates will also require some development of fabrication methods. In addition, some flow distribution and expansion joint tests will be required.

c. Refueling Machine.—Design development and testing of a suitable refueling mechanism will be required.

d. Fuel Element.—Tests of the fuel element assembly under various conditions of flow, temperature, and support will be required. It is anticipated that these tests will be out of pile. Major fuel element irradiation testing will be done in the prototype itself.

FIGURE 1.1—Concept No. 3A, boiling D₂O, pressure tube, direct cycle.

"A" - MOISTURE AT MOISTURE REMOVAL STAGE
 "B" - 1% STEAM WITH "A"
 "C" - MOISTURE HEATER EXTRACTION STEAM

200,000 KW UNIT - 1800 RPM
 T.C.D.F. - 38"
 750 PSIG
 SATURATED STEAM

GROSS GENERATOR OUTPUT	KW	220,059
AUXILIARY POWER @ 5.34 %	KW	11,745
NET SENT OUT	KW	208,314
STEAM RATE TO TURBINE	#/KWH	11.547
BTU/KWH TO TURBINE		11,981
NET STATION HEAT RATE	BTU/KWH	12,657
REACTOR MODERATOR HEAT LOSS %		7
PLANT HEAT RATE (100 % OPER. EFF.)	BTU/KWH	13,610
THERMAL EFFICIENCY	%	25.08

NOTE: P DENOTES PSIA

FIGURE 1.2—Performance diagram, concept No. 3A, boiling D₂O, pressure tube, direct cycle.

FIGURE 1.3—Reactor arrangement, prototype Title I design.

FIGURE 1.4—Composite flow diagram, prototype Title I design.

e. Component Tests.—There will be a number of component tests that will probably prove desirable during the course of the program. These cannot now be listed in detail, but none are expected to delay the prototype construction program.

G. POTENTIAL AREAS FOR IMPROVEMENT

The boiling D₂O pressure tube concept has excellent potential for future development. Among the development potentials S&L-NDA considered important are the following:

“1. The achievement of higher steam temperatures and pressures through the use

of internal insulation within the pressure tube to increase the allowable stress for the pressure tube.

“2. The substitution of H₂O (especially “sprayed” H₂O) for D₂O cooling, thereby reducing D₂O inventory, reducing D₂O inventory subject to contamination and losses, and decreasing the cost of the external system by eliminating fluid recovery systems.

“3. Superheating steam by passing the saturated steam generated back through superheating tubes within the reactor. Thus, more conventional powerplant steam conditions may be achieved within the reactor complex.”

Appendix 2—STATUS OF STUDY OF D₂O-MODERATED POWER REACTORS

E. I. DU PONT DE NEMOURS & CO., INC. (du Pont)

The program of the du Pont study of D₂O-moderated power reactors is outlined in this appendix. This study was undertaken in November 1956 at the request of the Atomic Energy Commission. A wide variety of reactors were evaluated in the study but only three different types are discussed in this report: (1) a boiling-D₂O-cooled, pressure tube reactor, (2) a liquid-D₂O-cooled, pressure vessel reactor, and (3) a liquid-D₂O-cooled, pressure tube reactor. These reactors were chosen from the du Pont study to show the variation of costs with capacity; they were also judged to possess the best scaleup characteristics of those considered. Because of the numerous similarities among these three concepts, all three are discussed as a group in this appendix rather than individually.

A. CONCEPT DESCRIPTION

Three reactor concepts are described in Table I; four designs, ranging in output from 100 eMW to over 400 eMW, were prepared for each concept. Thus, there are twelve reactor designs shown for the du Pont study.

All of the du Pont reactors are fueled with natural uranium metal. The designs shown in Table I include the following:

Type	Designation	Output (eMW)
Boiling D ₂ O, pressure tube-----	1K	100
Direct cycle-----	1K-200	200
	1K-300	300
	1K-430	430
Liquid D ₂ O, pressure vessel-----	1B	100
Hot moderator-----	1B-200	200
Indirect cycle-----	1B-300	300
	1B-400	400

Type	Designation	Output (eMW)
Liquid D ₂ O, pressure tube-----	1D	100
Indirect cycle-----	1D-200	200
	1D-300	300
	1D-460	460

Drawings of the reactor of largest capacity in each group are given in Figures 2.1, 2.2, and 2.3.

B. OBJECTIVES

The primary objective of the du Pont Power Reactor Study is to establish the technology required for successful design, construction and operation of a full-scale power reactor that is fueled with natural uranium and moderated with heavy water.

A second objective of the study is to survey the economic potential of the more favorable reactor concepts and to estimate the relationship between cost of power and electrical output.

C. WORK COMPLETED

1. General Research and Development

a. Physics Experiments

An experimental survey defined the nuclear parameters of a wide variety of lattices of natural uranium fuel elements in heavy water moderator. The type of fuel elements for which bucklings were measured include D₂O-cooled assemblies of uranium metal plates, uranium metal rods, uranium metal tubes, and uranium oxide rods, as well as gas-cooled assemblies of uranium metal plates. However, all of these experiments were made with mod-

erator temperatures below 95° C. Other important physics parameters of typical power reactor lattices of natural uranium in heavy water were also measured. These measurements included items such as the worth of control rods, the effect of H₂O contamination of the D₂O moderator, the fast fission factor, the neutron age, flux distribution, etc. In addition, correlations were derived from the experimental data for calculating resonance integrals and resonance escape probabilities for fuel elements of uranium oxide as well as of uranium metal. All of the measurements were made in either the Process Development Pile (PDP) or the Subcritical Experimental Facility (SE) at the Savannah River Laboratory.

All the measurements discussed in the preceding paragraph were made at temperatures below 95° C. In order to extend the range of the experimental physics work to moderator temperatures as high as 215° C the pressurized exponential facility (PSE) was built. The unit was put into operation during the first half of 1959. Nuclear parameters on a variety of lattices of natural uranium fuel elements are being measured at higher temperatures.

Among the important general physics calculations that were completed, other than those required for predicting the behavior of a particular power reactor lattice, were the following: (a) the effect of moderator temperature and fuel exposure on nuclear reactivity and core lifetime, (b) the variation of the thermal neutron reproduction factor with exposure and moderator temperature, (c) xenon transients, and (d) the response of various reactors to changes in temperature, reactivity or load.

b. Fuel and Materials

An extensive program has been in progress to fabricate low-cost fuel elements and components suitable for use in D₂O-cooled power reactors. Primary interest in the du Pont study has been on uranium metal fuel elements, but oxide elements are also being developed as an alternative to the metal fuel. The metal was selected in preference to the oxide as the

fuel for initial development, in spite of better irradiate stability and corrosion resistance of the oxide, because of the generally higher reactivity, lower cost and better known fabrication process of the metallic fuels.

The potentially low cost coextrusion process was applied by Nuclear Metals, Inc. (under a du Pont subcontract) to the manufacture of Zircaloy-2 clad tubes of U-2 w/o Zr. A "production" run of about a dozen tubes was successfully carried out. The feasibility of the manufacturing process was thereby proven. Numerous corrosion tests in hot water have been made of this fuel, with and without intentional defects. The fuel is deemed to be suitable from a corrosion viewpoint for a D₂O-cooled power reactor.

Potentially low cost methods of mechanical compaction of oxide powder to high density have been applied to the manufacture of rods and some tubes of uranium oxide fuel. Among the techniques used were swagging, rolling, extrusion, and explosive compaction.

Extruded zirconium and Zircaloy components are being fabricated by Harvey Aluminum Company. Very promising results were obtained in the manufacture of thin-walled, ribbed housing tubes of Zircaloy-2. Work was done at Nuclear Metals, Inc. on the manufacture of a metallurgical joint between Zircaloy and stainless steel by coextrusion.

One of the most important phases of the research and development program on fuels for D₂O-moderated power reactors involves the irradiation testing of prototype reactor fuel elements. The ability of uranium metal to adequately withstand long exposures to neutron irradiation was tentatively verified at low temperature. In order to irradiate significant numbers of full-size fuel elements at high temperature, the design and construction of the Heavy Water Components Test Reactor (HWCTR) was authorized.

c. Heat Transfer Experiments (Loops)

An experimental study of heat transfer burn-out of water-cooled tubular assemblies of the

type designed for the du Pont concepts of liquid-D₂O-cooled power reactors was completed at the Heat Transfer Research Facility of Columbia University. The range of variables of this study was limited to the particular range that is applicable to the du Pont power reactors. A study was also conducted of the vibration phenomenon that was encountered in some of the burnout experiments. The results of the experimental study were used to compute the safety factor on heat transfer burnout for the liquid-water-cooled assemblies of the power reactor designs and of the HWCTR.

d. Fluid Flow Experiments

Full-scale flow tests of the liquid-cooled fuel elements of the du Pont power reactors have been run at the Savannah River Laboratory. The corrosion, erosion and vibration characteristics, as well as the customary pressure-drop-versus-flow behavior, of these fuel elements were and are still being determined.

e. Coolant Chemistry

The condition of the water specified for the power reactor designs was duplicated in some of the fluid flow experiments mentioned above. The purpose of these experiments was to determine the corrosion properties of the coolant on fuel-element cladding, zirconium components, and other structural materials in the system. The lifetime and effectiveness of the resins intended for maintaining the purity of the coolant are also being determined in these fluid flow experiments.

f. Reactor Safety

Considerable theoretical study of the transient behavior and the responses of the reactor to unusual incidents (i.e., changes in reactivity, temperature, etc.) were studied, particularly for the hot moderator pressure vessel reactor and the HWCTR. It was shown that the hot moderator reactor and the HWCTR are to a great extent self-regulating and that such un-

usual incidents would cause no difficulty in control. A thorough analysis of the safety of the experimental reactor, the HWCTR, was completed.

g. Components and Auxiliary System

Detailed mechanical design of crucial components and auxiliary systems have been completed for the D₂O-cooled-and-moderated power reactors. A test program (See Section D, this appendix) has been devised for the testing of numerous important mechanical features.

2. Experimental Reactors

The Heavy Water Components Test Reactor (HWCTR) is a test facility in which as many as twelve prototype power reactor fuel elements will be irradiated simultaneously at high temperatures. Construction of this unit was authorized by the Atomic Energy Commission on November 6, 1958. The primary purpose of the HWCTR is the determination of the stability of various fuels under neutron irradiation. No electric power will be produced from this unit. The goal for startup for the HWCTR is early in 1961. At present (July 1959), about 45 percent of the firm design for the HWCTR is completed. Preliminary construction work has begun at the HWCTR site. The estimated cost of the facility is \$7,700,000.

a. Description, Purpose, Chronology, and Significant Design Parameters

The HWCTR is to be a thermal heterogeneous reactor that will be cooled and moderated with heavy water. The reactor will serve as a facility in which natural uranium fuel assemblies can be exposed to the same conditions of irradiation, temperature, pressure, heat flux, heavy water flow, and general environment as are expected in the du Pont designs of full-scale power reactors. Various types of fuel assemblies are under consideration for use in these reactors. It is expected that the first type of assembly that will be tested will consist of a tube of natural uranium-2 w/o zirconium clad with Zircaloy-2 and/or Zircaloy-4,

and surrounded by a ribbed, thin-wall Zircaloy housing tube. Heavy water will pass at velocities of about 10 to 20 feet/second through the coolant passages of the assembly. Twelve such natural uranium fuel assemblies will be tested simultaneously. These test assemblies will be located in the central part of the reactor. The outer part of the reactor core will contain 24 enriched uranium fuel assemblies that will provide the necessary reactivity for the small-diameter reactor.

The test reactor can be modified at some later date so that up to six in-pile loops may be installed. Two of these loops are authorized for construction and are being designed. These loops would be used for testing fuel assemblies of special design or for testing assemblies cooled with fluids other than liquid D₂O. Fuel design and installation of all loops represents an extension of the HWCTR program beyond the original scope of the project. However, the design of the test reactor must be suitable to accommodate such modifications when, and if, they are required.

The principal design and operating conditions for the test reactor are specified as follows:

Maximum power level, 61 megawatts.

Maximum D₂O outlet temperature, 285°C.

Normal D₂O outlet temperature, 240°C.

Minimum D₂O outlet temperature at maximum power, 225°C.

Temperature rise through reactor at maximum power, 27°C.

Design pressure, 1,500 p.s.i.g.

D₂O flow through reactor, 9,600 g.p.m.

Pressure drop through fuel, (estimated), 25 ft. of D₂O.

The following more detailed requirements were developed as the design study progressed:

The reactor vessel is to accommodate 10-foot long test fuel assemblies. These elements are shorter than those planned for the full-scale power reactor, but are certainly adequate to give reliable test results.

A total of 24 "driver" fuel assemblies of enriched uranium alloyed with zirconium will

be used. These assemblies, which will be 10 feet long, are to be arranged on a 20-inch radius, surrounding the test fuel assemblies. The driver tubes will be housed in Zircaloy-2 and/or Zircaloy-4 tubes and will contain boron-stainless steel target elements. The purpose of the target elements is to provide a burnable poison in the reactor to compensate for the burnup of the enriched uranium during the reactor cycle and thereby permit more constant reactivity and longer exposure cycle.

A total of 6 safety rods on a 12-inch radius, 6 instrument positions on an 8.085-inch radius, 12 control-rod positions on a 15½ inch radius, and a cluster of 6 control rods on the central axis of the core are to be provided. Monitor pins are to be located at the outlet of each fuel assembly to permit measurement of the flow and temperature rise in the coolant.

The reactor will have a removable top head. The charging and discharging of fuel assemblies after removal of the head will be accomplished by means of a transfer coffin working through an indexing top shield plug in the top of the reactor.

b. Cross-Section View and Flow Diagram

Figures 2.4 and 2.5 are, respectively, an arrangement of the HWCTR reactor vessel and a flow diagram for the facility.

c. Operating Experience

None.

3. Power Demonstration Reactors

a. Description

All the reactors described in Section A, this appendix, are of the full-scale power demonstration type.

b. Purpose

See Section B, this appendix.

The purpose of these reactor designs is merely to extend the knowledge of the economics of D₂O-moderated power reactors. No authorization for construction of these reactors was given, nor is it anticipated.

c. Chronology

None. It is expected, however, that these reactors could be built by 1965.

d. Significant Design Parameters

See Table I of the body of the report.

e. Cross-Section Views

See Figures 2.1, 2.2 and 2.3.

f. Operating Experience

None.

D. WORK UNDERWAY**1. General Research and Development and Experimental Reactor**

Major emphasis in the experimental program is now focussed on development of data necessary for the successful design, construction, and operation of the HWCTR. However, much of the work done for the HWCTR is applicable to the large-scale reactors as well. The general purpose of the program is still evident in the work under way, to extend the technical knowledge of D₂O-moderated power reactors.

a. Physics Experiments

Experimental evaluations of the following items concerning the HWCTR are now in progress in the PDP, PSE and SE of the Savannah River Laboratory:

- (1) Control and safety rod systems.
- (2) Flux distributions.
- (3) Neutron economy.

These PDP experiments are at room temperature but additional measurements are being made at D₂O temperatures as high as 215° C in the PSE.

A program is also in progress to extend the survey of the nuclear parameters of oxide lattices. This will be done in the PDP.

A sizable program is underway to extend the measurements of lattice parameters to mod-

erator temperatures as high as 215° C. The pressurized exponential facility (PSE) was built for this purpose. The same types of measurements as were mentioned in Section C. 1.a are planned.

b. Fuel and Materials

The fabrication program mentioned in Section C. 1.b is continuing, with the purpose of improving the quality and lowering the costs of manufacturing fuel and components.

The irradiation tests that are now in progress or that are definitely planned are listed in Table II.

In addition, the program for testing metallic fuels is currently being expanded to include the following proposed tests:

(1) Irradiation of U-2 w/o Zr alloy in different conditions of heat treatment to determine whether or not a significant change in microstructure affects irradiation stability.

(2) Irradiation of unalloyed uranium of both high and low carbon content. The effect of grain size will also be investigated. Data previously obtained at Savannah River Plant with unalloyed uranium tubes indicate that the unalloyed metal behaved more stably under irradiation than did the U-2 w/o Zr alloy in the diffusion heat-treated condition.

(3) Irradiation of dilute alloys of uranium containing aluminum, silicon, or molybdenum to further investigate the effects of grain size, microstructure, and creep properties on irradiation stability.

(4) Other systems requiring more development work are also being considered. Possible irradiation candidates in this category include U₃Si alloy and U-UO₂ mixtures.

The fabrication of some of the fuel elements for these tests has already begun.

c. Heat Transfer Experiments (Loops)

Work is being extended at Columbia University in order to study the heat transfer burnout characteristics of fuel elements in wa-

ter with net steam formation. These data are required for defining the operating limits of the boiling-D₂O-cooled power reactors.

d. Fluid Flow Experiments

Fluid flow experiments are being expanded to include the static and cyclic testing of pressure seals, access hole closures, and pressure tube connections, as well as fuel element assemblies of the HWCTR and the full-scale reactors.

e. Coolant Chemistry

Further study of resin lifetime and impurity-removal-effectiveness will be made in the continued fluid flow experiments.

f. Reactor Safety

A complete reactor safeguard study was done for the HWCTR. Review will be made of the study during the design and construction of the facility.

g. Components and Auxiliary Systems

Mechanical testing of monitoring devices, safety and control rod mechanisms, charge-discharge machinery, seals, etc., will continue. Further work will also be done on the corrosion characteristics of various components.

2. Power Demonstration Reactors

The above discussion, in Section D. 1 also applies to the large-scale power demonstration reactors.

E. CONSTRUCTION AND OPERATION SCHEDULE

None for power demonstration reactors.

F. INHERENT PROBLEMS AND LIMITATIONS

The main problems and possible limitations of the du Pont designs of large-scale power reactors are summarized as follows:

1. Physics

Does the lattice have sufficient reactivity to permit long exposure of fuel and allow for adequate control of the reactor at design power level? This question will be answered for the three du Pont reactor concepts by the experimental physics program underway. For the hot moderator pressure vessel reactor, PSE measurements will be made at moderator temperatures up to 215° C. For both the pressure tube reactors and the pressure vessel reactor, measurements will be made in the SE and PDP, with mockups of the various reactor lattices.

2. Fuel

The adequacy of metallic fuel elements to withstand long exposure to neutron irradiation under conditions comparable to those designed for the power reactor, must be demonstrated. Otherwise, oxide fuel elements may have to be substituted for the metal. This problem is common to the boiling-D₂O as well as the liquid-D₂O-cooled reactors. This is probably the most important problem in the du Pont designs.

3. Heat Transfer

The burnout characteristics of the liquid-D₂O-cooled assemblies have been defined but further work is required to define this limit for the boiling-D₂O-cooled assemblies.

4. Coolant Stability and Temperature

No problem is envisioned here.

5. Components

Tests of the crucial mechanical features of both the pressure tube and pressure vessel reactors are required. Such tests are planned and should resolve the engineering problems in the designs. However, one of the greater problems exists in the pressure tube reactor, as com-

pared to the pressure vessel; that is, the fabrication of suitable Zircaloy pressure tubes and joints between these tubes and other materials.

G. POTENTIAL AREAS FOR IMPROVEMENT

Among the items that may be developed and that can affect the cost of power from these reactors are the following:

1. Metallic fuel elements that can withstand long exposure satisfactorily. This will permit a fuel shuffling or relocation scheme within the reactor and thereby lower the fuel cost.

2. Development of high-strength, low-parasitic-absorption pressure tubes.

3. Means of incorporating nuclear superheat in the reactor.

4. Reductions in the fabrication costs of fuel and zirconium components.

TABLE II
LIST OF IRRADIATION TESTS

Number	Irradiation Site	Core Material	Clad Material	Core Length	Temperature Condition	In Progress
Uranium Metal Tubes:						
2.....	SRP	U-2 w/o Zr.....	Zircaloy-2.....	10'.....	Low	Now
1.....	NRU	U-2 w/o Zr.....	do.....	10'.....	High	3/59
1.....	VBWR	U-2 w/o Zr (enriched).	do.....	3'.....	High	Now
Uranium Metal Slugs:						
8.....	SRP	U-2 w/o Zr.....	do.....	10''.....	High	Now
Uranium Oxide Slugs:						
4.....	MTR	Fused UO ₂	Stainless Steel.....	10''.....	High	Now
Various Fuels:						
Several specimens.....	HWCTR	U Metal.....	Zircaloy.....	10'.....	High	1961
		UO ₂	do.....

FIGURE 2.1—Pressure vessel reactor. Liquid D₂O cooled; hot moderator (400 eMW).

GENERAL NOTES

COOLANT SUPPLY PRESSURE	813 PSIG
HOUSING DESIGN PRESSURE	1010 PSIG
INLET TEMPERATURE	212° C
OUTLET TEMPERATURE	252° C
NO OF FUEL TUBES	330
NO OF CONTROL RODS	30
NO OF SAFETY RODS	30
LATTICE SPACING	9½"
D ₂ O VOL. COOLANT SYSTEM	13,900 LBS.
D ₂ O VOL. MODERATOR TANK	202,200 LBS.
RE CORE WEIGHT	46,200 LBS.

FIGURE 2.2—Pressure tube reactor. Liquid D₂O cooled; cold moderator (460 eMW).

FIGURE 2.4—Arrangement of HWCTR reactor vessel.

FIGURE 2.5—Cooling system flow diagram for the HWCTR.

Appendix 3—STATUS OF PROGRAM ON D₂O-COOLED-AND-MODERATED POWER REACTORS, NUCLEAR POWER DEMONSTRATION REACTOR (NPD), AND CANADIAN DEUTERIUM URANIUM REACTOR (CANDU)

Atomic Energy of Canada, Ltd.

A. CONCEPT DESCRIPTION

See Table I of the body of this report and Figures 3.1 and 3.2 of this appendix.

A major difference between the Canadian pressurized-liquid-D₂O power reactor concepts and those designed in the United States is that the Canadian reactors are horizontal whereas U.S. reactors are vertical structures. The main reason for the horizontal feature of the Canadian NPD and CANDU is to permit easy bidirectional refueling while the reactor is at power. Bidirectional fueling is the feeding of fuel slugs in opposite directions in adjacent channels. The use of short slugs pushed through the reactor a step at a time results in the discharging of fuel, all of which has reached the same exposure level.

B. OBJECTIVE

The purpose of the Canadian power reactor program is the development of low-cost thermal-electric plants for the rapidly expanding industrial areas of Canada where hydroelectric resources are tapped nearly to their limit and fossil fuel reserves are inadequate (specific technical objectives are discussed below).

C. WORK COMPLETED

1. General Research and Development

The Canadian NRX and NRU heavy water, natural uranium metal, test reactors have provided much information on this general type

of reactor. Research and development work specifically undertaken for the power reactor program has been completed to the following extent. In the succeeding outline, the small reactor refers to the NPD and the large reactor refers to the CANDU.

a. Physics Experiments

Experiments were performed in the zero-energy heavy water reactors (ZEEP) at Chalk River with natural and slightly enriched uranium metal and with uranium oxide in various fuel geometries and lattice configurations. The purpose of these studies was to obtain optimum lattice designs for large and small power reactors.

b. Fuel and Materials Properties

Many irradiation tests were carried out on Zircaloy-clad UO₂ fuel elements in the NRX and elsewhere, largely under conditions which will prevail in the Canadian power reactors. Oxide center temperature, oxygen-uranium ratio, oxide density, and fission gas release were related as a result of this work and a suitable Zircaloy-clad uranium oxide fuel element was developed for use in the Canadian power program. Significant progress was also made in proving the adequacy of Zircaloy-2 for high-pressure in-pile pressure tubes.

c. Heat Transfer Experiments (Loops)

Numerous heat transfer tests were run in the many in-pile loops at Chalk River. A maxi-

mum safe heat flux has been established for the oxide rod-type element operating in heavy water. A high-pressure boiling-water facility for fundamental study of boiling systems was also built. Electrically heated elements were studied in out-of-pile loops at Canadian General Electric's Peterborough Plant to provide refined heat transfer data for the small and large reactor designs.

d. Fluid Flow Experiments

Considerable work on hydraulic studies for the Canadian power reactor program was done at the above-mentioned Peterborough Plant, where diversified equipment was erected including loops operating with ordinary water up to 100 atmospheres and 310° C. at high flow rates. Mockup fuel elements have been photographed under various power reactor operating conditions to study the flow patterns and enable design refinement.

e. Coolant Chemistry

Much of the coolant chemistry experience which is being incorporated in the first Canadian power reactor designs was obtained in operation of the NRX and NRU. Further work on coolant chemistry and corrosion is continuing.

f. Reactor Safety

Experiments which were carried out to confirm the safety aspects of the Canadian power reactor schemes included a series of tests in which pressure tubes were ruptured to determine the effect on the enclosing calandria tubes and adjacent tubes. Proposed control systems and related circuitry were also tested for operability and reliability.

g. Components and Auxiliary Systems

The most important work in this area involved the construction of a prototype refuel-

ing machine. Mockups or prototypes of all important components, the performance of which cannot be evaluated adequately on a theoretical basis, will be tested before use in the first plant.

2. Experimental and Power Demonstration Reactors

To date, no reactors in the Canadian power reactor program have been completed.

D. WORK UNDERWAY

1. General Research and Development

Because design of neither the NPD nor the CANDU is complete, there is no distinct line between work that has been completed and work that is still underway. There is, therefore, some overlapping of completed and incomplete work in the following discussion.

a. Physics Experiments

A critical assembly consisting of uranium oxide pellets in aluminum tubes is currently being assembled for studies of the Nuclear Power Demonstration Reactor (NPD-2). This work will be carried out in the ZED-II at Chalk River, a facility of considerably larger diameter than the ZEEP.

b. Fuel and Materials Properties

Irradiation of prototype and developmental fuel elements, including swaged oxide rods, is continuing with the objectives of increasing fuel life, reducing cost, and proof-testing the designs to be used in the initial reactor. The development of Zircaloy pressure tubes continues in an effort to assure maximum strength at minimum cost and neutron loss. A Zircaloy tube has been in use in the NRX since 1957 and several more have subsequently been installed in the NRU. The performance of these tubes will be evaluated.

c. Heat Transfer, Fluid Flow, Coolant Chemistry, and Reactor Safety Experiments

These studies are continuing at Peterborough to refine the large and small reactor designs. A large-scale mockup with two pumps, eight loops, and a heat exchanger is being constructed at Peterborough to enable further confirmation of power reactor design data, especially for the full-scale plants. Various components, including several Byron Jackson main coolant pumps and the full-scale prototype refueling machine, are being tested for the first plant.

2. Power Demonstration Reactors

a. Description

(1) NPD-2, Nuclear Power Demonstration Station, is a small power-producing plant under construction on the Ottawa River at Des Joachims, 16 miles upstream from the Chalk River site. This plant is the scale-down prototype of the "CANDU" plant described below. The reactor is a horizontal pressure-tube type capable of being refueled on-power bi-directionally.

(2) CANDU, the Canadian Deuterium Uranium reactor, is a full scale power-producing plant that is now in the preliminary design stage and essentially of the same design as the NPD-2. However, its power rating is 200 eMW compared to 19 eMW for the NPD.

b. Purpose

(1) NPD-2 is intended to determine the technical feasibility of the CANDU concept. Among the specific items of importance are the refueling technique, the use of Zircaloy pressure tubes and their associated connections and fittings, the determination of heavy water losses in such plants, the testing of fuel elements, the determination of load factor for such plants, and the training of personnel for subsequent large-scale plants.

(2) CANDU is intended to establish on a full scale the economics of a reactor plant of this concept.

c. Chronology of Canadian Program

Early 1954: Commenced studies, preliminary designs, and evaluations of heavy water, natural uranium power reactors.

Early 1955: Design began on NPD-1, which was a concept similar to NPD-2 but with vertical-axis pressure vessel instead of horizontal pressure tubes.

Late 1955: Large reactor study started.

Early 1957: NPD-1 construction commenced.

Mid 1957: NPD-1 construction halted for re-evaluation of pressure tube versus pressure vessel concepts. Design of NPD-2 began.

Early 1958: Preliminary studies of large-scale (CANDU) reactor complete.

Mid 1959 (present): Concrete for NPD-2 about 30 percent poured; NPD-2 design is 85 percent complete; NPD-2 project is considered to be 50 percent complete. CANDU studies continue.

d. Significant Design Parameters

See Table I of the body of this report.

e. Cross Section Views and Flow Diagrams

The concept and flow diagram for the Canadian power reactors are shown in Figures 3.1 and 3.2, respectively.

E. CONSTRUCTION AND OPERATION SCHEDULE

End 1960—All equipment installed in NPD-2.

Early 1961—NPD-2 construction complete.

Mid 1961—NPD-2 in operation.

Mid 1962—CANDU design expected to be complete and construction started.

1965-1966—CANDU expected to be in operation.

F. INHERENT PROBLEMS AND LIMITATIONS

1. Reactivity and Physics

NPD-2 has a positive coolant void coefficient, meaning that a loss or boiling of the coolant adds reactivity and tends to increase the power. The protective system guards against this by providing shutdown at a rate greater than that necessary to overcome the positive effect. Any serious change in coolant conditions will initiate a shutdown. In the remote event that the shutdown initiation is not achieved through some peculiar malfunction of the control system, there is no automatic or self-regulating protection of the reactor. However, the control system employs triplicated instruments in dual-coincidence circuits to provide a high degree of reliability.

2. Fuel and Materials

A very remote possibility exists in this reactor concept of the exothermic reaction of the Zircaloy tubes and sheaths with water or steam. The Canadians believe that it is almost impossible for the Zircaloy in NPD-2 to reach the temperature necessary for the reaction while the Zircaloy is still in contact with water or steam.

Hydrogen embrittlement of the Zircaloy sheaths has been observed in some tests and could cause cracking and subsequent excessive release of fission gas products into the coolant. It is believed that close control of Zircaloy impurities and fabrication methods will minimize this problem. Similarly, the Zircaloy pressure tube design necessary for reasonable neutron economy does not allow much leeway on strength. Also, the effects of irradiation on Zircaloy strength are not fully known. Studies

are continuing with the purpose of determining the in-pile behavior of various zirconium alloys, particularly Zircaloy-2.

3. Heat Transfer

Heat transfer is limited by the restriction on boiling in this pressurized water concept, as described above. Furthermore, the heat flux is limited by the maximum operating temperature of the fuel element core, which has been established at a level several hundred degrees below the melting point of uranium dioxide.

4. Coolant Stability and Temperature

As in pressurized light water reactors, the coolant is relatively stable. However, the temperature is limited by the pressure which can be safely maintained in the primary system.

5. Components

Components and accessories which are potential trouble spots include the mechanical joint connecting the pressure tubes to the end fittings, the mechanical end fittings themselves, the refueling machine, and the "leak proof" primary coolant pumps. All these items could cause excessive D₂O losses and mechanical difficulties in operation.

G. POTENTIAL AREAS OF IMPROVEMENT

Neither the NPD-2 nor the CANDU is completed and in operation. Therefore, potential areas of improvement are not yet clearly defined. One improvement which is not included in the NPD-2 design, but which is contemplated for the CANDU, is the facility for one refueling machine to load completely independent of the other assigned to unloading.

FIGURE 3.1—Sketch of Nuclear Power Demonstration (NPD) reactor.

FIGURE 3.2—Flow diagram of Nuclear Power Demonstration (NPD) reactor.

Appendix 4—STATUS OF PROGRAM ON D₂O-COOLED-AND-MODERATED POWER REACTOR

Carolinas Virginia Nuclear Power Associates (CVNPA)

A. CONCEPT DESCRIPTION

See Table I of body of report.

The proposed reactor operates with a thermal neutron spectrum and is basically a D₂O-moderated-and-cooled, pressure tube reactor, which requires enriched fuel for small units, but which may operate with natural uranium in the large (>200 eMW) sizes. Zircaloy-2 clad fuel in the form of UO₂ rods is contained in vertical, U-type pressure tubes of Zircaloy-2, and is cooled by forced convection with D₂O at 1500 p.s.i.a. The pressure tubes are insulated internally from the high temperature coolant by three thin layers of stagnant D₂O and are cooled externally by the moderator which is maintained at atmospheric pressure and approximately 155° F. This concept incorporates steam generators in which the reactor heat is transferred to the light water secondary coolant loop. Saturated steam from the boilers at about 605 p.s.i.a. is superheated in a fossil-fueled superheater before entering the turbine. A gross efficiency for electrical power generation of 30 percent (including superheat) is anticipated.

B. OBJECTIVES

The prime objectives of this reactor concept are (1) the extension of the pressurized water type of reactor to sizes that may be beyond the range of practicable pressure-vessel construction, and (2) the utilization of materials and methods that show promise in the attainment of low-cost nuclear power.

C. WORK COMPLETED AND UNDERWAY

1. General Research and Development

In general, work in this category to date has consisted primarily of analytical studies, consultations with AEC contractors, and literature searches to aid in the resolution of alternative design features in arriving at a reference design. Progress on specific portions of this work is outlined below.

a. Physics Experiments

A critical facility is being designed specifically for this project and is to be built at the Westinghouse Reactor Evaluation Center. Experiments are planned to start approximately May 1960. In these tests the nuclear parameters pertinent to the CVTR (Carolinas-Virginia Tube Reactor) design will be evaluated. The facility will consist of full-scale mockup of the core of the CVTR, utilizing about four metric tons of UO₂ as fuel. Two enrichments (1.1 and 2.0% U-235) of the UO₂ will be studied. Reactivity measurements will be made for a number of sizes of clusters of fuel rods, varying the pitch of the fuel rods within the clusters, and varying the D₂O-to-UO₂ volume ratio between 10 and 20.

b. Fuel and Materials Properties

Tests are currently underway on the fabrication of UO₂ rods, clad with Zircaloy-2, by extrusion and swaging techniques. Brazing and welding tests are being conducted to evalu-

ate methods of joining Zircaloy-2. Various methods for joining Zircaloy-2 to stainless steels are being investigated. Plans have been made to perform the following tests:

(1) Corrosion of clad fuel, both defected and undefected.

(2) In-pile irradiation tests of clusters of fuel rods in the Westinghouse Test Reactor to begin after January 1960.

(3) Strength and corrosion tests of Al_2O_3 -Al alloys as alternative pressure tube materials.

(4) Continued tests of fuel fabrication techniques.

(5) Tests of the compatibility of the joints and materials of the primary system with coolant and moderator under appropriate conditions of temperature and pressure and under irradiation.

c. Heat Transfer Experiments (Loops)

An existing Westinghouse loop is being modified to permit the following high pressure, high temperature tests:

(1) Pressure tube thermal performance and corrosion life.

(2) Pressure tube fittings, joints and components.

(3) Thermal tests simulating various coolant and moderator flows.

These tests are planned for the period July 1, 1959 to Dec. 31, 1959.

d. Fluid Flow Experiments

Another existing Westinghouse loop is being modified to permit the following hydraulic tests under low pressure and low temperature coolant conditions:

(1) Moderator circulation patterns to be studied visually.

(2) Hydraulic tests of fuel assembly components.

Upon completion of these tests, a full-scale fuel assembly will be inserted into the high pressure loop (described in "c" above) for hydraulic tests under typical reactor operating conditions.

e. Coolant Chemistry

Tests have been initiated to evaluate ion-exchange media for coolant and moderator purification; to date, both high temperature inorganic compounds and low temperature organic resins have been tested on simulated impure coolant. Tests are also being conducted on filter media and on resins for deoxygenating the coolant.

f. Reactor Safety

Analytical studies on various aspects of reactor safety (reactivity changes and transient load conditions) are being performed.

g. Components and Auxiliary Systems

In addition to the large number of analytical studies underway on the various phases of the primary and secondary coolant circuits, the transients anticipated in the reactor and steam generation coupled systems, the pressurizer requirements, etc., a number of tests are planned on the metallurgical behavior of pressure tubes, instruments to measure dimensional changes and creep remotely, methods of minimizing D_2O leakage from the various components of the reactor complex, control rod materials both for strength and corrosion resistance, control rod drive mechanisms, and on flow, activity and temperature monitoring instruments.

2. Power Demonstration Reactors

Almost all of the research and development effort described in section C.1 is being conducted primarily to clarify the best design for the CVTR, the prototype reactor for the Carolinas-Virginia's concept of a large-scale liquid- D_2O -cooled reactor.

A 270 eMW design was described in the CVNPA proposal to the AEC, August 23, 1957. In that proposal a rough estimate was made of the cost of power based on an early design concept; it was 9.8 mills/kwh. All subsequent effort has been directed toward the attainment of a reference design for the CVTR.

a. Description

See Table I of body of report and Figures 4.1 and 4.2 of this appendix.

The CVTR is a vertical pressure-tube power reactor in which the Zircaloy-2 pressure tubes are thermally insulated from the high temperature coolant by three nominally 0.064-inch thick, stagnant layers of D₂O, defined by thin (0.021-inch) Zircaloy baffle tubes mounted concentrically within the pressure tube. Within each 3.53-inch I.D. pressure tube, a cluster of 19 fuel rods is suspended. Each rod is 0.50-inch outside diameter and 8 feet in active length. The rods consist of enriched UO₂ pellets (or extruded rods) clad with Zircaloy-2 of 0.032-inch thickness. The rods are spaced in the cluster by helically wound, Zircaloy wires appropriately selected to provide a rod pitch of 0.60 inch.

The core of the CVTR consists of 84 pressure tubes placed on 8.0- by 6.5-inch rectangular lattice spacing. Adjacent pairs of pressure tubes are connected at the bottom of the reactor via a 180° return bend and thus, form U-tube assemblies. Coolant flows in at the top of one of the pair of pressure tubes, down through the fuel, around the 180° bend, up through the second fuel cluster and out through the top of the second pressure tube. The pressure tube pairs are connected via jumpers (2-inch Schedule 80 pipe) to the main coolant inlet and outlet headers (8-inch Schedule 100 pipe). Two 3,800 g.p.m., 500-foot head, centrifugal, canned motor pumps in parallel serve to circulate the coolant in the primary circuit.

The 42 U-tubes are immersed in the moderator which is contained by a vessel 123 inches in diameter and 140 inches in overall height. The moderator is maintained at atmospheric pressure and at a temperature between 130° F and 180° F. The moderator absorbs approximately 9.5 percent of the reactor thermal power—6.5 percent from gammas and neutrons and 3.0 percent from heat loss through the pressure tubes—and is cooled and circulated through a system separate from the high temperature

coolant. D₂O reflectors, 2 feet thick in the radial direction and 1.5 feet thick at the top and bottom of the core, surround the fueled lattice.

Coolant at 1,500 p.s.i.a. and 555° F leaves the reactor and flows to a heat exchanger at a rate of 3.3×10^6 lbs./hr. In the heat exchanger, the primary coolant loses 56 MW of heat, leaving at 505° F, while saturated steam is generated in the secondary circuit at a rate of 202,000 lbs./hr. and at a pressure of 605 p.s.i.a. Steam from the CVTR is to be fed to existing turbines of the Parr, S.C., power plant; it is planned to superheat the steam from the CVTR to 700° F in auxiliary, conventional superheating equipment prior to feeding to the turbine at this temperature and a pressure of 415 p.s.i.a.

b. Purpose

The purpose of the CVTR is to demonstrate the feasibility and to evaluate the economic potential of the pressure tube, pressurized-water concept. It is designed to serve as a prototype for large-scale (i.e., 200 eMW) power reactors.

It is hoped that all features of the concept can be proven via operation of the CVTR, except that of operation with natural uranium fuel.

c. Chronology

- 1957 and 1958—Analytical design studies small scale testing.
- 1959—Advanced design studies; mechanical, materials and fuel fabrication testing.
- 1960—Critical experimental facility; firm nuclear design of CVTR; initial construction.
- 1961—Full-scale construction effort.
- 1962—Startup and operation of CVTR; design studies of large pressure-tube power reactor.

d. Significant Design Parameters

See Table I of body of report.

e. Cross-Section View and Flow Diagram

Figure 4.1, Reactor Plant Vertical Section.
Figure 4.2, Primary Coolant Flow Diagram.

f. Operating Experience

None.

D. WORK UNDERWAY

This category is discussed in Section C, this appendix.

E. CONSTRUCTION AND OPERATION SCHEDULE

See Figure 4.3.

F. INHERENT PROBLEMS AND LIMITATIONS**1. Reactivity and Physics**

The problems in the physics of the Carolinas-Virginia reactor concept arise primarily from a lack of knowledge of D_2O - UO_2 lattices. Since physics data for D_2O - UO_2 lattices are available only for the cold, clean condition

- (a) in substitution lattices (UO_2 replacing U-metal elements),
- (b) for natural uranium oxide, and
- (c) over a limited range of lattices spacings and clumping ratios,

it is only with reluctance that estimates of reactivity in the hot, clean and hot, dirty conditions are made, in view of the extent of extrapolation required. The critical facility, however, will permit firm estimates to be made for the CVTR and similar cores. Estimates of core lifetimes for various enrichments as a function of lattice spacing and cluster size must await results from the critical experiment.

There appear to be no inherent limitations of this system at this time. Results of critical experiments may show that it will not be pos-

sible for large reactors of this type (>200 eMW) to operate with natural uranium oxide as fuel. This statement becomes more probable if insurance rates and insurability are to be based on adherence to the ASME Code for in-pile structural materials, e.g., the pressure tubes. Thus, it is safe to say that the largest single problem facing the CVTR designers is that of reducing the amount of parasitic (structural) materials in the core.

Analyses will also be required of the transient behavior of the reactor in responding to reactivity and load changes.

2. Fuel and Materials

Basically, the use of UO_2 as fuel simplifies the problems associated with fuel element dimensional stability and integrity of the sheathing material. However, whether UO_2 should be compartmented to prevent thermal ratcheting (between UO_2 and Zircaloy-2) and to minimize the effects of pellet disintegration under thermal stress and cycling remains an unsolved problem. Furthermore, detailed knowledge of the maximum allowable UO_2 temperature is not available, nor is it now known that centerline melting of UO_2 is hazardous.

Problems associated with the clad UO_2 fuel in rod form are:

- (a) Is brazing of Zircaloy-2 end plugs to the tubular sheath a satisfactory closure?
- (b) What hazards are associated with relatively large pellet-to-sheathing clearances?
- (c) Are brazed or welded fittings acceptable for fuel rod support?

Problems associated with the materials appear to be:

- (a) What are reasonable values for the high temperature strength of Zircaloy-2? Clarification of A.S.M.E. Code requirements is needed.
- (b) Are Zircaloy-2 welds or Zircaloy-to-stainless-steel joints acceptable in regions of significant neutron and gamma flux levels?

(c) How does Zircaloy-2 hydriding take place? How can it be avoided?

(d) How susceptible is Zircaloy-2 to corrosion and erosion in forced convection, high temperature D_2O ?

(e) Can Zircaloy-2 be reliably formed by upsetting processes and can such pieces be irradiated without loss of strength?

(f) What is the effect of coolant quality on material life? If pH control or burnable poisons are used, do they seriously affect the life of welded, brazed or otherwise fabricated pieces?

Answers to most of these questions are being sought in current and future testing studies. Although none of the problems listed appears to be an inherent limitation, each of them obviously affects significantly the attainment of economic power via this concept.

3. Heat Transfer

Although there are no inherent limitations that arise from the heat transfer, and hydraulic mechanisms proposed for the CVTR, a number of problems exist.

(a) Methods of reducing the heat loss to the moderator from the pressure tubes (currently about 3% of reactor power) are sought.

(b) Methods of improving the mixing of coolant within the pressure tubes (to minimize hot spots and reduce power limiting hot-spot factors) are being investigated.

(c) Means of eliminating the requirement that the thermal baffles be discharged with each fuel bundle are being sought to minimize fueling costs.

(d) Improved methods of joining the U-bend and stainless-steel pressure tube extensions to the Zircaloy-2 pressure tubes would permit design simplifications.

4. Coolant Stability and Temperature

The relatively high stability of D_2O and the ease of purification (via ion-exchange media)

as well as the ease of recombination of the dissociation gases prevent serious problems or limitations from arising.

5. Components

The problems associated with the components of the reactor complex are relatively minor and no inherent limitations have been found. Studies are proceeding to determine the economic optimum designs for the various components including:

(a) The moderator tank; aluminum or stainless steel?

(b) The steam generator; inconel tubes appear more attractive than either stainless or mild steel, since greater corrosion and erosion resistance permits higher coolant velocity resulting in higher heat transfer coefficients and hence, smaller surface requirements.

(c) Piping sizes; high coolant velocities (~ 38 f.p.s.) appear optimum based on the balance between pumping power costs and the combined cost of D_2O inventory and stainless steel within the process piping.

G. POTENTIAL AREAS OF IMPROVEMENT

1. Core Structural Materials

Should the proposed research and development effort uncover suitable means of increasing the high-temperature strength of Zircaloy-2, evolve reliable brazing or other methods of joining Zircaloy-2 to stainless steel, and essentially eliminate D_2O loss from the system, significant improvements can be anticipated in the approach to economic nuclear power. The reduction of the quantity of structural material (mostly Zircaloy-2) in the core is the primary concern of the CVTR designers. Not only will reactivity of the lattice improve with removal of parasitic absorbers, but space limits will be eased as well, permitting more flexibility in the core design.

2. Critical Experiments

Should the measurements of lattice parameters obtained in the critical experiments permit the *validation* of methods of computation of reactivity throughout the irradiation cycle, improved cores will be designed. Perhaps a

more economic core (than the current, uniformly enriched one) can be devised by use of enriched "spikes" or a "seed" region with the major portion of the fuel composed of natural UO_2 . Eventually, the enriched portion of the core could be made of recycled plutonium with an improvement in the cost of power.

PLANT CONTAINER ELEVATION "A-A"

FIGURE 4.1—Plant arrangement of Carolinas-Virginia Tube Reactor (CVTR).

RAIN

VENT & DRAIN
ITEM AND
DECONTAMIN
ON SYSTEM

NG SYSTEM

ITER

OWN LINE

0 SCRAM

E STEAM
EATER

PRESSURE AT
TURBINE
THROTTLE

FIGURE 4.2—Primary coolant system flow diagram for Carolinas-Virginia Tube Reactor (CVTR).

CRITICALITY TEST

Request Allocation of UF_6
 Start UF_6 Delivery
 UO_2 Delivery
 Pellet Delivery
 Element Manufacturing
 Critical Experiments

CONTAINMENT VESSEL

Complete Reference Design
 Complete Design and Place Steel Order
 Fabricate and Ship Steel
 Foundations
 Interior Concrete Structures
 Steel Shell Erection and Testing

MAIN COOLANT SYSTEMS COMPONENTS-ORDERS PLACED

Steam Generator
 Pumps
 Valves
 Piping

FUEL ASSEMBLIES FOR POWER CORE

Request Allocation for UF_6
 UO_2 Delivery
 Pellet Delivery
 Fabrication of Fuel Elements
 Charging Fuel to Reactor

PLANT TESTS

Hydrostatic Test
 Preliminary System Test
 Construction Complete—Startup
 Low Power Operation

■ = Target Dates

— Interval during which indicated operations take place.

FIGURE 4.3—Tentative CVTR construction and operation schedule.

Appendix 5—STATUS OF PROGRAM ON THE SODIUM DEUTERIUM REACTOR (SDR)

Nuclear Development Corporation of America (NDA)

A. CONCEPT

See Table I of the body of this report.

The SDR concept combines the high thermal performance of sodium as a coolant with the high neutron efficiency of D_2O as a moderator in an attempt to bring about economic nuclear power, through the achievement of high steam cycle efficiency and the use of natural uranium fuel. The advantages of the sodium coolant include a low operating pressure even at high temperatures, which permits the use of thin-walled housing tubes made of moderately high cross section materials and makes it possible to generate steam at modern power plant conditions, with materials of current technology. The objective of the NDA study was to explore the economic potential of natural uranium SDR power plants, and to outline the effect of variation of significant design parameters on energy cost.

Among several mechanical arrangements considered, a vertical calandria design with "straight-through" fuel tubes received the major attention. This design, illustrated in Figure 5.1, includes a vertical aluminum calandria containing nonpressurized D_2O . The fuel, in the form of clustered rods, is contained within vertical thin-walled housing tubes which pass through the calandria tubes. The sodium coolant flows within the housing tubes in a single upward pass, entering and leaving the housing tubes through flexible small-diameter tubing (pigtailed) connecting to the main sodium headers. A barrier tube is placed between the housing tube and the calandria tube to provide backup separation between the sodium and D_2O , in the event of simultaneous

leaks in a calandria tube and its associated housing tube. Nitrogen gas between the housing tube and the barrier, and between the barrier and the calandria tube, provides effective thermal insulation between the sodium and the D_2O .

External to the reactor, an intermediate heat exchanger is used to transfer heat from the primary sodium coolant to a secondary, non-radioactive, sodium system, which in turn transfers its heat to the steam generator, as shown in Figure 5.2.

B. OBJECTIVES

The primary objectives of the SDR program are:

(a) To demonstrate the feasibility of separation of sodium and heavy water in the SDR reactor.

(b) To establish the potential of the SDR concept as an economic power producing system.

(c) Complete the preliminary design of a 10 eMW power plant useful for the commercial generation of electrical power and capable of contributing primary design and operating data for the natural uranium SDR, (i.e., pilot plant).

C. WORK COMPLETED

1. General Research and Development

a. Physics Experiments

The nuclear characteristics of natural uranium SDR-type lattices were studied in a three-part program that consisted of:

(1) A short experimental program, carried out in the Process Development Pile of the Savannah River Laboratory.

(2) The development of theoretical calculation techniques specifically for SDR-type reactors.

(3) A survey of experimental measurements of D_2O lattices and a comparison of these with NDA calculation methods.

The results of this program, although not conclusive, indicated that values of reactivity obtained by engineering methods were no more than 0.5 to 1.0 percent optimistic in k_{eff} for the important survey designs. A number of measures are available for increasing reactivity margins; these measures were reviewed and the increase in the cost of power resulting from them was found generally to be less than 0.5 mills/kwh.

b. Fuel and Materials Properties

SDR-type reactors were designed utilizing three types of fuel: UO_2 , metal and UC (uranium carbide). The primary effort in this category was directed to the computation of numerous reactor designs based on each type of fuel. In these computations, the properties and behavior of the fuels were obtained from literature searches.

Considerable effort was expended in establishing the compatibility of the sodium coolant with the structural and cladding materials employed in the several reactor designs. Additional work was performed in the study of the reaction rate between high-temperature sodium and low-temperature, liquid H_2O and the vapors of both H_2O and D_2O .

Tests were performed in which jets of water and sodium were directed at specimens of barrier tube and calandria tube materials. These tests have demonstrated that a single aluminum barrier will maintain the separation of sodium and water in the event that leaks develop in either or both of the primary containers holding the fluids. The barrier will

not be penetrated by leaks of sodium at temperatures up to the melting point of the aluminum. At sodium temperatures above this melting point, a jet may pass through the barrier, but the aluminum calandria tube backed (and cooled) by water will resist the jet for considerable lengths of time.

c. Heat Transfer and Fluid Flow Tests (Loops)

An experimental loop, the SDR Mockup, was operated with sodium circulation for more than 4,000 hours, of which more than 3,000 hours were with both sodium and water. The test program consisted of four major activities:

(1) Thermally cycling the apparatus between $750^\circ F$ and $950^\circ F$ on an 8-hour cycle.

(2) Thermally shocking the apparatus so that a $50^\circ F$ temperature difference existed at the most critical fuel tube section.

(3) Testing leak detection equipment by simulating leaks of both sodium and water.

(4) Continuation of normal operations at high load factor.

The loop was operated successfully through all tests with no evidence of test section failure. In addition, the leak detection equipment operated successfully with rapid responses to simulated leaks of sodium, water and air.

Thus, this program demonstrated that the SDR system can be operated reliably and with no major difficulty.

d. Coolant Chemistry

Studies were conducted to determine the purity required of the sodium coolant for compatibility with in-pile materials. An upper limit of 10 p.p.m. of oxygen was established as being satisfactory for Zircaloy-2. The use of dry nitrogen as a gas blanket over all surfaces of exposed sodium was demonstrated as a sound operating practice. Conventional methods of maintaining high sodium purity in the presence of stainless steel and Zircaloy-2 were proven satisfactory in the SDR designs.

e. Reactor Safety

(1) Inherent Safety Features

D₂O moderation results in a long prompt neutron lifetime and slow responses to reactivity disturbances, affording time for corrective action to be taken by the control system.

The SDR is a low pressure system of high thermal capacity. Low pressures in the reactor coolant and moderator minimize stored energy in the fluids and therefore no large forces are present to magnify system failures. Sudden loss of coolant by flashing is impossible.

(2) Design Safety Features

The choice of the calandria-housing tube concept for the SDR allows a complete mechanical independence of the hot sodium and cold D₂O systems. The stainless steel sodium system is all-welded, with a minimum of critical welds. D₂O system design is based on proven technology developed by Savannah River, Chalk River, and other installations.

Complete separation of sodium and D₂O results from the use of an independent metal barrier between the fluids. This mechanical barrier concept was carried throughout the reactor core and external systems. Failure of at least three separate, independent, metal walls must occur before the fluids can mix.

All rooms where sodium is present are flooded with dry nitrogen, maintained at less than 1 percent oxygen concentration, to prevent sodium combustion. An oxygen concentration of 4 percent is required to support combustion of sodium, or hydrogen that might form in a sodium-water reaction.

Adequate expansion volumes and vent devices insure that shock waves do not arise and that hydrogen generated in a sodium-water reaction can be relieved.

Two independent leak detection systems, based on both liquid and vapor detectors are provided to check for sodium leakage. Each fuel channel is monitored with detectors for gross liquid spillage.

An important feature of the SDR is that circulation of the blanket gas through the barrier annuli can be used as effective shutdown cooling to remove reactor afterheat.

In addition to these features, the following conventional safety provisions are included: use of emergency battery and diesel power supply; use of two half-capacity sodium pumps to maintain adequate cooling should either pump fail; and use of normal interlocks and fail-safe electronic design.

(3) Safety Analysis

An analysis of possible reactivity incidents showed that no major excursions can occur because no mechanism exists whereby large excess reactivities can be introduced into the core.

Incidents that were analyzed include:

- (a) Cold, clean start-up accident.
- (b) Total loss of sodium coolant.
- (c) Failure of fuel, barrier and calandria tubes with mixing of sodium and water in the calandria.

In each case, it is found that no shock waves result and that the containment building can withstand the most severe accident and prevent gross fission product release to the atmosphere.

f. Components and Auxiliary Systems

(1) Leak Detection

Tests were performed that provided data for specifying the sensitivity required of the sodium vapor detection system. Several commercially available instruments meet these requirements.

Instrumentation was developed which provides not only indications of liquid sodium leaks, but also provides information of system malfunction.

(2) Fuel-Coolant Tube Closures

A test program was initiated to develop a gas-tight mechanical closure consisting of a metal O-ring seal, acting in conjunction with a mechanical loading and locking device.

Tests were conducted to determine helium leakage at conditions simulating those in an SDR, (e.g., remote insertion and removal) the presence of equilibrium vapor pressure of sodium at 1,100° F, ambient temperatures of 150° F to 400° F and 20 p.s.i. pressure difference, and both thermal and mechanical cycling.

(3) *Calandria Tube-Tube Sheet Joints*

Typical expansion joints, capable of accommodating thermal expansions and stresses between the calandria tube and tube sheet, were tested to determine the suitability of the seals and moving parts under simulated reactor conditions of mechanical and thermal cycling. Four different joints were tested—three bellows assemblies and one sliding seal assembly.

Further tests are required before selection of the most suitable design can be made.

(4) *Remote Repair*

Tools were designed, constructed, and successfully tested over working distances approximating full-scale reactor dimensions, demonstrating the feasibility of remotely repairing and replacing a fuel-coolant tube (i.e., housing) in an SDR. These tools and accessories functioned well over a working distance of 30 feet.

(5) *Mechanical Joints*

Commercially available mechanical joints for connecting sodium piping were tested and evaluated. They remained leak-proof for more than 2,000 hours of continuous operation and 3,300 thermal cycles.

FIGURE 5.1—Elevation layout of a single-region 200-eMW sodium-deuterium reactor.

FIGURE 5.2—Schematic of a single-region 200-eMW sodium-deuterium reactor.

FIGURE 5.3—Construction and operation schedule for the SDR program.

Appendix 6—REFERENCES

A. Sargent & Lundy and Nuclear Development Corporation of America

SARGENT & LUNDY and NUCLEAR DEVELOPMENT CORPORATION OF AMERICA, *Heavy Water Moderated Power Reactor Plants. Design Study. Sec. I, Comparison and Evaluation*, USAEC Report TID-8503 (Part 1), July 1959.

SARGENT & LUNDY and NUCLEAR DEVELOPMENT CORPORATION OF AMERICA, *Heavy Water Moderated Power Reactor Plants. Design Study. Sec. II, Preliminary Plant Designs*, USAEC Report TID-8503 (Part 1), July 1959.

SARGENT & LUNDY and NUCLEAR DEVELOPMENT CORPORATION OF AMERICA, *Heavy Water Moderated Power Reactor Plants. Design Study. Sec. III, Appendixes*, USAEC Report TID-8503 (Part 1), July 1959.

SARGENT & LUNDY and NUCLEAR DEVELOPMENT CORPORATION OF AMERICA, *Heavy Water Moderated Power Reactor Plant. Preliminary Design of the Prototype Plant. Sec. I, Prototype Title 1 Design Description*, USAEC Report TID-8503 (Part 2), July 1959.

SARGENT & LUNDY and NUCLEAR DEVELOPMENT CORPORATION OF AMERICA, *Heavy Water Moderated Power Reactor Plants. Preliminary Design of the Prototype Plant. Sec. II, Prototype Title 1 Design Drawings and Outline Specifications*, USAEC Report TID-8503 (Part 2), July 1959.

SARGENT & LUNDY and NUCLEAR DEVELOPMENT CORPORATION OF AMERICA, *Heavy Water Moderated Power Reactor Plant. Preliminary Design of the Prototype Plant. Sec. III, Prototype Preliminary Safety Report*, USAEC Report TID-8503 (Part 2), July 1959.

SARGENT & LUNDY and NUCLEAR DEVELOPMENT CORPORATION OF AMERICA, *Heavy Water Moderated Power Reactors—Additional Design Studies and Development Program, Design Studies*, USAEC Report SL-1653 (Part 3), June 19, 1959.

B. E. I. du Pont de Nemours & Co., Inc.

D. F. BARCOCK, E. I. DuPont de Nemours & Co., *Heavy Water Moderated and Cooled Power Reactors*, A/CONF.15/P/610, prepared for the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, September 1958.

D. F. BARCOCK, Design of a 100-MW Electrical Power Plant Using Heavy Water as Reactor Moderator, Chapter 5 of *Solid Fuel Reactors*, edited by J. R. DIETRICH and W. H. ZINN, 1958.

F. E. DRIGGERS and D. S. ST. JOHN, Physics of Natural Uranium, Heavy Water Reactors, Chapter 5 of *Solid Fuel Reactors*, edited by J. R. DIETRICH and W. H. ZINN, 1958.

G. DESSAUER, E. I. DuPont de Nemours & Co., *Physics of Natural Uranium Lattices in Heavy Water*, A/CONF.15/P/590, prepared for the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, September 1958.

A. H. FUTCH, Fast Fission Effect in Lattices of Natural Uranium and Heavy Water, *Nuclear Sci. and Eng.*, 5: No. 1: 61-67 (1959).

R. C. HOLMES et al., *A Preliminary Evaluation of Gas Cooling of Power Reactors Moderated by Heavy Water*, USAEC Report DP-307, E. I. DuPont de Nemours & Co., August 1958.

R. L. MENEGUS, *Burnout of Heating Surfaces in Water*, USAEC Report DP-363, E. I. DuPont de Nemours & Co., May 1959.

D. RANDALL and D. S. ST. JOHN, Xenon Spatial Oscillations, *Nucleonics*, 16: No. 3: 82-86, 129 (1958).

D. S. ST. JOHN et al., *Preliminary Hazards Evaluation of the Heavy Water Components Test Reactor (HWCTR)*, USAEC Report DP-383, E. I. DuPont de Nemours & Co., May 1959.

J. W. WADE, Neutron Age in Mixtures of D₂O and H₂O, *Nuclear Sci. and Eng.*, 4: No. 1: 12-24 (1958).

E. I. DU PONT DE NEMOURS & Co., Heavy Water Moderated Power Reactor Studies Quarterly Progress Report for the Periods: *February, March, and April 1957*, USAEC Report, DP-232; *May, June, and July 1957*, USAEC Report DP-245; *August, September, and October 1957*, USAEC Report DP-265; *November 1957 through January 1958*, USAEC Report DP-285; *February, March, and April 1958*, USAEC Report DP-295; *May, June, and July 1958*, USAEC Report DP-315; *August, September, and October 1958*, USAEC Report DP-345; *November 1958 through January 1959*, USAEC Report DP-375.

C. Atomic Energy of Canada, Ltd.

ATOMIC ENERGY OF CANADA LIMITED, *The Canadian Study for a Full-Scale Nuclear Power Plant*, Report AECL-557, January 1958.

ATOMIC ENERGY OF CANADA LIMITED, *Atomic Power Symposium Held at Chalk River May 4 and 5, 1959*, Report AECL-799 A and B.

W. B. LEWIS, *First-Day Summing-Up (Paper 1-6) of Atomic Power Symposium 4-5 May 1959*.

W. B. LEWIS, Atomic Energy of Canada, Ltd., *Canada's Steps Towards Nuclear Power*, A/CONF.15/P/183, prepared for the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, September 1958.

H. A. SMITH et al., Atomic Energy of Canada, Ltd., *A Study of a Full-Scale, Uranium and Heavy Water Nuclear Power Plant*, A/CONF.15/P/208, prepared for the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, September 1958.

I. N. MACKEY, Canadian General Electric Company, Ltd., *The Canadian NPD-2 Nuclear Power Station*, A/CONF.15/P/209, prepared for the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, September 1958.

Design Description—Canada's First Nuclear Power Station. A booklet prepared by Civilian Atomic Power Department, Canadian General Electric Company, Limited, Peterborough, Ontario, Canada, Report NPD-2, September 1958.

ATOMIC ENERGY OF CANADA, LTD., personal communications, June 1959.

D. Carolinas Virginia Nuclear Power Associates

Proposal to the United States Atomic Energy Commission by Carolinas Virginia Nuclear Power Associates, August 23, 1957.

CAROLINAS VIRGINIA NUCLEAR POWER ASSOCIATES, INC., *Carolinas Virginia Tube Reactor Project—Work Programs Prepared by Atomic Power Department, Westinghouse Electric Company*, USAEC Report CVNA-21, April 20, 1959.

CAROLINAS VIRGINIA NUCLEAR POWER ASSOCIATES, INC., *Quarterly Progress Report for the Period January 1 to March 31, 1959*, USAEC Report CVNA-22, May 20, 1959.

WESTINGHOUSE ELECTRIC CORP., *Carolinas Virginia Nuclear Power Associates CVTR*, Project Progress Report for Feb. 17, 1959 to Mar. 31, 1959, USAEC Report WACP-1148.

CAROLINAS VIRGINIA NUCLEAR POWER ASSOCIATES, INC., *CVTR Project Monthly Progress Report for Apr. 1959*, USAEC Report WACP-1183.

CAROLINAS VIRGINIA NUCLEAR POWER ASSOCIATES, personal communication, June 1959.

E. Nuclear Development Corporation of America

NUCLEAR DEVELOPMENT CORPORATION OF AMERICA, *Preliminary Safety Analysis of the Sodium Deuterium Reactor SDR*, (10 MWe Chugach) Report NDA 84-14, March 6, 1959.

NUCLEAR DEVELOPMENT CORPORATION OF AMERICA, *Natural Uranium Sodium-Deuterium Reactors*, Report NDA-84-15, Vol. I, Feb. 28, 1959; Vol. II, March 15, 1959.

NUCLEAR DEVELOPMENT CORPORATION OF AMERICA, *10 MWe SDR Design Report*, Report NDA-84-16, Vol. I, February 28, 1959; Vol. II, Parts 1 and 2, March 31, 1959.

W. L. BROOKS et al., *Reactivity Studies on Natural Uranium-D₂O Lattices*, USAEC Report NDA-84-17, Nuclear Development Corp. of America, March 31, 1959.

C. BOLTA, *SDR Mockup*, USAEC Report NDA-84-18, Nuclear Development Corp. of America, April 22, 1959.

L. CORRISIN et al., *Sodium-Water Reaction Rate Studies*, USAEC Report NDA-84-19, Nuclear Development Corp. of America, May 15, 1959.

H. BELOFSKY et al., *Testing of Sodium Pipe Joints and Development of Remote Repair Tools for the SDR*, USAEC Report NDA-84-20, Nuclear Development Corp. of America, May 22, 1959.

C. BOLTA et al., *Design Development Tests of Some Components for the 10 MWe SDR*, USAEC Report NDA-84-21, Nuclear Development Corp. of America, May 25, 1959.

NUCLEAR DEVELOPMENT CORPORATION OF AMERICA, personal communication, June 1959.