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ATOMIC ENERGY OF CANADA LIMITED

L'ÉNERGIE ATOMIQUE DU CANADA LIMITÉE

AN OVERVIEW OF THE POTENTIAL OF THE CANDU REACTOR AS A THERMAL BREEDER

by

J.B. SLATER

Paper based on a presentation to the IAEA Consultants Meeting on the Status and Prospects of Thermal Breeders, Vienna, 15-17 September, 1976

Chalk River Nuclear Laboratories

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Aperçu du potentiel du réacteur CANDU comme réacteur surrégénérateur thermique

Par

J.B. Slater

Résumé

Le Canada a développé un réacteur modéré à l'eau lourde pour la production d'électricité appelé CANDU. Ce réacteur utilise l'eau lourde à la fois, comme modérateur et comme caloporteur, et il emploie de l'oxyde d'uranium naturel comme combustible. Cependant, une caractéristique importante du concept de base de la filière CANDU, est qu'il peut être amené à employer différents combustibles (uranium, plutonium ou thorium). On traite dans ce rapport de l'emploi du thorium comme combustible dans le concept actuel des réacteurs CANDU. On analyse le bilan neutronique du coeur du réacteur et on évalue le potentiel de développement d'un réacteur surrégénérateur thermique. On indique, pour conclure, que bien que le cycle de thorium à équilibre auto-suffisant semble faisable, il est peu probable qu'on puisse développer un cycle de combustible significativement surrégénérateur si l'on veut conserver la capabilité de fonctionnement actuel des réacteurs CANDU et leur coût en capital.

L'Energie Atomique du Canada, Limitée
Laboratoires Nucléaires de Chalk River
Chalk River, Ontario

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ABSTRACT

Canada has developed a heavy water moderated reactor for production of nuclear-electric power called the CANDU. This reactor uses heavy water (deuterium oxide) as both coolant and moderator, and natural uranium oxide as fuel. However, an important feature of the basic CANDU concept is that it can evolve to use different fuels (uranium, plutonium or thorium). This paper is concerned with the use of thorium as a fuel in the existing CANDU concept. The neutron balance of the reactor core is analyzed and an assessment is made of the potential for development of a thermal "breeder" reactor system. It is concluded that while the SSET cycle (i.e. self-sufficient equilibrium thorium cycle) appears feasible, there is little potential for developing a significant "breeding" fuel cycle if current reactor operating capability and capital costs are to be maintained.

Reactor Physics Branch
Chalk River Nuclear Laboratories
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AN OVERVIEW OF THE POTENTIAL OF THE CANDU REACTOR AS A THERMAL BREEDER

by

J. B. Slater

1. INTRODUCTION

Canada has developed a heavy water moderated reactor called the CANDU for production of nuclear-electric power. This reactor uses heavy water (deuterium oxide) as both coolant and moderator, and natural uranium oxide as fuel. This has enabled Canada to develop a nuclear industry without the high initial capital expense of uranium enrichment and fuel reprocessing plants, although D_2O production plants are required.

An important feature of the basic CANDU concept is that it can evolve to use different coolants (light water or organic liquids) and different fuels (uranium, plutonium or thorium). Development of the use of the alternative coolants will permit reductions in nuclear plant capital cost. However, this paper is concerned with the use of thorium as a fuel in the existing CANDU concept and the potential for development of a thermal "breeder" reactor system.

2. THE EXISTING CONCEPT

Main features of the current reactor core design are illustrated on Figure 1. Heavy water moderator is held in the calandria tank which is penetrated by horizontal calandria/pressure tube combinations, in a square array, which contain the fuel. The fuel is cooled by pressurized heavy water (typically at about 10 MPa) which is circulated through the pressure tubes. In-core instrumentation and reactivity devices for reactor power shaping and control are contained in other tubes which penetrate the calandria tank, generally in the vertical direction (Figure 2). Similarly, two independent shut-down systems are provided.

The heat removed by the primary coolant is used to produce steam to drive the turbine by means of an indirect cycle.

The fuel (Figure 3) is natural uranium oxide contained in Zirconium alloy sheaths. The basic fuel unit is a short (500 mm) fuel bundle consisting typically of either 28-(Pickering) or 37-elements (Bruce, 600 MW(e)* designs) Current CANDU designs require a loading of 12 bundles per channel. The fuel design is relatively simple, consisting of only six components and the short length facilitates easy handling and part-channel, on-power refuelling.

The concept has been successfully used in several power plants, among them the four-unit 2000 MW(e) Pickering Generating Station (Reference 1) near Toronto. A list of operating reactors and those under construction is given in Table 1. Generally, operating experience has been satisfactory, being characterized by high availability and low operating and fuelling costs (e.g. Reference 2).

---

* e - <u>e</u>lectrical power

TABLE 1

CANADIAN NATURAL URANIUM HEAVY WATER POWER REACTORS IN OPERATION, UNDER CONSTRUCTION, COMMITTED OR PLANNED

NAME[1]	LOCATION	TYPE[2]	POWER MW(e) NET	NUCLEAR DESIGNER[3]	DATE OF FIRST POWER
NPD	Ontario	PHW	22	AECL & CGE	1962
Douglas Point	Ontario	PHW	208	AECL	1967
Pickering A	Ontario	PHW	514 x 4	AECL	1971–73
Gentilly 1	Quebec	BLW	250	AECL	1971
KANUPP	Pakistan	PHW	125	CGE	1971
RAPP 1	India	PHW	203	AECL	1972
RAPP 2	India	PHW	203	AECL	1976
Bruce A	Ontario	PHW	745 x 4	AECL	1976–78
Gentilly 2	Quebec	PHW	600	AECL	1980
Cordoba	Argentina	PHW	600	AECL	1979
Pickering B	Ontario	PHW	514 x 4	AECL	1981–83
Point Lepreau	New Brunswick	PHW	600	AECL	1981
Wolsung 1*	Korea	PHW	600	AECL	1980
Bruce B**	Ontario	PHW	750 x 4	AECL	1983–86
Darlington**	Ontario	PHW	800 x 4	AECL	1985–88

[1]NPD — Nuclear Power Demonstration
KANUPP — Karachi Nuclear Power Project
RAPP — Rajasthan Atomic Power Project

[2]PHW — *Pressurized Heavy Water Coolant*
BLW — Boiling Light Water Coolant

[3]AECL — Atomic Energy of Canada Limited
CGE — Canadian General Electric Company Limited

* — Under Negotiation
** — Planned

The problems encountered in the Pickering-3 and -4 units due to cracking of some of the zirconium-niobium pressure tubes have been successfully resolved. The Pickering units are now back at full power and zirconium-niobium has been retained as reference pressure tube material for future units.

3. THORIUM FUEL

Because of the superior nuclear properties of U233 in a thermal neutron spectrum, the use of thorium as a fuel for the CANDU reactor has been studied for many years (e.g. References 3, 4, 5). Recently, Atomic Energy of Canada Limited has carried out a general survey of the economic and performance characteristics of these cycles (References 6, 7, 8). Some typical results from this survey (particularly Reference 7) will now be briefly discussed. These particular results apply to cycles using fissile plutonium (supplied from a natural uranium fuelled CANDU-PHW), plus thorium with uranium cycle in a CANDU-PHW. All reactor units are a 1200 MW(e) size with a fairly standard lattice design, and use 37-element oxide fuel bundles in 10.34 cm (4", nominal inner diameter) pressure tubes. The design power for the maximum rated channel is 6.5 MW (thermal). The important parameters for the natural uranium fuelled reactor are a fuel burnup of 7.5 MWd/kg U with spent fuel containing 2.7 g fissile plutonium per kg. Note that in the general survey studies no allowance was made for fissile material losses in fuel fabrication and reprocessing.

Figure 4 shows equilibrium burnup per pass and conversion ratio for the thorium fuelled CANDU-PHW as a function of fissile Pu content of the feed fuel (Th + recycled U + Pu). Also shown are system total unit energy cost and system equilibrium uranium requirements.

The conversion ratio is defined as the ratio of total fissile atom production to total fissile atom destruction. As expected, the burnup increases with increasing fissile Pu content in the feed fuel. As the burnup increases the conversion ratio decreases and the system equilibrium uranium requirements increase. The total unit energy cost has an optimum, but the curve is quite shallow.

Figure 5 gives exactly the same information as Figure 4 but in a slightly different form. The characteristics have been cross-plotted against system equilibrium uranium requirements to stress that aspect of the results. For zero system equilibrium uranium requirements the burnup is finite and the energy cost is reasonable. This we call a self-sufficient equilibrium thorium (SSET) cycle since at equilibrium no fissile Pu topping is required for the feed fuel (only recycled uranium). Hence at equilibrium no natural uranium reactors are needed in the system, which can operate with only an external feed of thorium.

For all these calculations it has been assumed that the uranium has been recycled many times and an "equilibrium" isotopic composition has been established, as given below.

U composition	232 :	Trace
	233 :	61%
	234 :	23%
	235 :	6%
	236 :	9%
	238 :	Trace.

4. POTENTIAL AS A THERMAL BREEDER

These survey results indicate that by using thorium fuel in a typical current CANDU lattice design, it should be possible to achieve a "self-sufficient" cycle with a fuel burn-up in excess of 10 MWd/kg HE*.

To assist in considering whether it is possible to develop a "true" breeding fuel cycle (i.e. producing significantly more fissile material than is consumed) a typical neutron balance has been derived for the SSET and is given in Table 2. It can be seen that the balance is dominated by neutron absorption and production in Th-232 and the uranium isotopes. An increase in the production of U-233 is obviously a question of "neutron economy", i.e. reducing non-productive absorptions so that additional neutrons can be captured in Th-232. The full potential increase in conversion ratio could be obtained if items 2, and 6-10 of Table 2 were all negligible and this would result in a value of $\sim$1.3. Obviously only part of this potential can be realized, as discussed below. Major constraints underlying the assessment of potential improvement are the requirements that changes should not adversely affect either capital costs or the current operating and load manoeuvering capabilities of the reactor.

---

*HE - Heavy Element

TABLE 2

NEUTRON BALANCE FOR CANDU CORE AT EQUILIBRIUM

a) Self-sufficient equilibrium thorium cycle, total U/Th = 2.6 wt %

b) Average fuel discharge irradiation = 10 MWd/kg HE

c) Average irradiation of fuel in core = 5 MWd/kg HE

d) Normalized to 1 absorption in U-233

	Absorptions	Current CANDU Design
1.	Th-232	1.03
2.	Pa-233	0.02
3.	U-233	1.00
4.	U-234	0.08
5.	U-235	0.10
6.	U-236 + higher actinides	0.01
7.	Fission Products	0.11
8.	Coolant, Moderator	0.04
9.	Fuel Sheathing and Structural Materials	0.07
10.	Leakage, Control and Power Shaping	0.09
11.	Total	2.55
	Production	
12.	U-233	2.29
13.	Th-232	0.03
14.	U-235	0.21
15.	Other	0.02
16.	Total	2.55

$$\text{Conversion Ratio} = \frac{\text{U233 + U235 production}}{\text{U233 + U235 destruction}} = \frac{1.03 - 0.02 + 0.08}{1.00 + 0.10} \approx 1.0$$

The discussions of potential increases in conversion ratio are grouped under the item numbers used in Table 2.

4.1 Items 2, 6, 7

Neutron captures in Pa-233, U-236 and fission products are parasitic, i.e. producing neither significant fissions nor further fissile material. The Pa-233 absorptions are particularly important since they also deny the equivalent formation of U-233 through radioactive decay.

One method of increasing neutron economy would be to reduce the average fuel specific rating and hence neutron flux level. This reduces the equilibrium levels of both Pa-233 and Xe-135 (which accounts for ~50% of the total fission product absorptions). A reduction of ~40% in rating results in ~0.04 increase in conversion ratio. However, the effect is non-linear and the first 10% reduction results in significantly less than 0.01 increase.

An alternative approach makes use of the on-power refuelling capability of the CANDU reactor. Fuel is discharged from the core part-way through its burnup cycle; a significant fraction of the Pa-233 is allowed to decay to U-233; the fuel is then recharged to the reactor to complete its burnup duty. The potential improvement in conversion ratio is estimated to be 0.005-0.01.

Both methods have drawbacks in other areas. Fuel rating reduction also implies higher reactor capital and fuel inventory costs and mid-cycle fuel removal increases both operational and fuel inventory costs.

4.2 Item 8

Approximately 50% of the absorptions in the coolant and moderator are caused by the light water component. Although the D_2O is maintained at high purity (~ 99.75% average) further improvement to the range 99.90-99.95% appears feasible, with little increased expense. The associated improvement in conversion ratio is 0.01-0.015.

4.3 Item 9

As a result of the continual program on improvements in methods of design (Reference 12) and nuclear materials (Reference 10), some further reduction in absorptions by fuel sheathing and structural materials is to be expected. In the longer term, development of laser isotope separation methods (Reference 11) could lead to further reductions through the use of structural materials enriched in low absorption isotopes. Associated potential conversion ratio improvements lie in the range 0.01 (near-term) to 0.04 (long-term).

4.4 Item 10

Current designs of CANDU reactors mainly use removable absorber for power shaping, control and power manoeuvres. As discussed in Reference 7, there is some scope for alternative methods of achieving power shaping with reduced neutron wastage. Potential reductions of associated neutron loss are in the range 10-20% with a corresponding increase in conversion ratio of 0.01-0.02.

The potential improvements are listed in Table 3 and the total range is +0.035 to +0.125. However, two further points should also be considered in this evaluation. The effect of fissile material losses during fabrication and reprocessing are equivalent to an effective reduction in conversion ratio, roughly on the scale, 1% loss = 0.015 reduction. It has been assumed that fissile losses can be kept to below 1%. Secondly, there are uncertainties in both basic nuclear data (e.g. as discussed in Reference 9 for U-233 data and fission product poisoning) and physics calculations and a minimum uncertainty of $\pm$ 0.03 must be assigned to conversion ratio estimates at this time.

The potential improvement, over the current CANDU design, is relatively small ($\sim$2.5%) at the low end of the range. This is perhaps not too surprising since the concept of neutron economy has always been a keystone in the design evolution. Further significant improvement is possible but only at the expense of either increased capital and operating costs (e.g. fuel derating) or on the assumption of technological "break-through" (laser isotope separation). Moreover, it is not obvious that further improvements in neutron economy would be used necessarily to increase conversion ratio. Improvements could be used to increase fuel burnup (0.01 in conversion ratio $\equiv$ 2 MWd/kg HE in burnup) and hence reduce fuel cycle costs and overall external fissile inventory requirements. This latter factor may be more important in a rapidly expanding nuclear power system.

TABLE 3

POTENTIAL IMPROVEMENTS IN CONVERSION RATIO

Base Conversion Ratio	~ 1.0
Reduction in Fuel Rating	0 to +0.04
Mid-cycle Fuel Removal/Replacement	+0.005 to +0.01
Maintain Higher D_2O Purity	+0.01 to +0.015
Improved Nuclear Materials	+0.01 to +0.04
Improved Power Shaping	+0.01 to +0.02
Total Potential Improvement	+0.035 to +0.125
Effect of Fissile Losses during Refabrication	~ -0.01
Uncertainties in Base Conversion Ratio	±0.03
Potential Conversion Ratio	~1.025 ± 0.03 to ~1.115 ± 0.03

In the light of the above, it is concluded that the realistic, near-term potential for the CANDU reactor is the SSET cycle in which there is no requirement for an external source of fissile material other than that required for the initial core and approach-to-equilibrium inventories.

5. SUMMARY

SSET cycles in 1200 MW(e) CANDU reactor units, achieving burnups of at least 10 MWd/kg HE per cycle, appear feasible provided fabrication and reprocessing losses of fissile material can be kept below 1% and careful attention is paid to neutron economy.

Achievable cumulative uranium requirements for indefinite operation of such cycles are (Reference 7):

i) $\sim$2 Mg/MW(e) if fissile plutonium produced in a natural uranium CANDU is used for initial fissile material requirements, and

ii) $\sim$1 Mg/MW(e) if highly enriched U-235 is used for initial fissile material requirements.

Data for other CANDU fuel cycles are summarized in Table 4.

There appears little potential for developing a significant "breeding" fuel cycle if current reactor operating capability and capital costs are to be maintained.

TABLE 4

PRELIMINARY REFERENCE CANDU-PHW CASES.

Assumptions: *37-Element UO_2 or ThO_2 Bundles in 10.34cm (nom) Channels at 28.6 cm Square Pitch.*
Avg. Specific Power: U cases: .0234 (MW/kg HE); Th Cases: .0263 (MW/kg HE)
Net Station Efficiency = 29.2%

CHARACTERISTIC		Natural Uranium	1.2% Enriched Uranium	Pu Recycle	Pu Topping			U-235 Topping		
					Th High Burnup	Th Inter-med.	Th Self-suff	Th High Burnup	Th Inter-med.	Th Self-suff
Identification										
Burnup (MWd/kg HE)		7.5	20.8	18.0	37.2	18.7	10.0	37.4	19.5	10.0
Feed Fuel		Nat. U	1.2% En.U	Nat. U + RecyclePu		(Th + Recycle U + Pu)			(Th + Recycle U + U-235)	
Effective Conversion Ratio[1]					.88	.96	1.00	.88	.96	1.00
Equilibrium Net Feed Rates for 1 GW(e) Unit Operating at 80% Load Factor										
Fissile Pu	(Mg/a)	-	-	-	0.1345	0.0535	0	-	-	-
U-235	(Mg/a)	-	-	-	-	-	-	0.1336	0.0513	0
Natural Uranium	(MgU/a)	133.4	-	55.6	-	-	-	-	-	-
1.2% Enriched Uranium	(MgU/a)	-	48.1	-	-	-	-	-	-	-
Thorium[3]	(MgTh/a)	-	-	-	26.9	53.5	100.1	26.7	51.3	100.1
Equilibrium Net Production Rates for 1 GW(e) Unit Operating at 80% Load Factor										
Fissile Pu	(Mg/a)	0.360	0.158	-	-	-	-	-	-	-
"Inventories"[2] for 1 GW(e) Unit (80% Load Factor)										
Delay Time = 1 year										
Fissile Pu	(Mg)	-	-	0.74	3.74	4.02	4.93	-	-	-
U-235	(Mg)	-	-	-	-	-	-	3.48	3.68	4.46
Natural Uranium	(MgU)	139.7	-	101.0	-	-	-	-	-	-
1.2% Enriched U	(MgTh)	-	97.2	-	-	-	-	-	-	-
Thorium[3]		-	-	-	78.6	91.9	115.2	78.5	90.8	115.2
Delay Time = 1.5 year										
Fissile Pu	(Mg)	-	-	0.80	4.07	4.61	5.98	-	-	-
U-235	(Mg)	-	-	-	-	-	-	3.77	4.19	5.43
Natural Uranium	(MgU)	173.0	-	114.9	-	-	-	-	-	-
1.2% Enriched U	(MgU)	-	109.3	-	-	-	-	-	-	-
Thorium[3]	(MgTh)	-	-	-	85.3	105.3	140.2	85.2	103.6	140.2

NOTES

[1] Average value for equilibrium fuelling conditions including an allowance for 0.5% fissile losses during fuel reprocessing.

[2] "Inventories" includes initial in-core inventory and additional "approach-to-equilibrium" requirements above those determined by the equilibrium fuelling rate. The inventory is a function of delay time, i.e. minimum time between discharge of recycle material from reactor and subsequent recharge.

[3] Assuming no recycle of thorium

REFERENCES

1. "Pickering Generating Station", Nuclear Engineering International, 15, 169, p.501-516 (1970).

   McINTYRE, H.C., Natural-uranium Heavy-water Reactors, p. 17, Scientific American, 233, 4, 17 (October 1975).

2. L.W. WOODHEAD and L.J. INGOLFSRUD, "Performance of Canadian Commercial Nuclear Units and Heavy Water Plants", presentation to European Nuclear Conference, April 21-25, 1975, Paris, France.

3. G.A. PON, W.B. LEWIS, L.R. HAYWOOD, D.B. PRIMEAU, G.J. PHILLIPS and E.E. MERLO, "Prospective D_2O Moderated Power Reactors", Proceedings of the 3rd International Conference on Peaceful Uses of Atomic Energy, 5, 333 (1964).

4. W.B. LEWIS, "How Much of the Rocks and Oceans for Power? Exploiting the Uranium-thorium Fission Cycle", Atomic Energy of Canada Limited, Report AECL-1916 (1964).

5. W.B. LEWIS, M.F. DURET, D.S. CRAIG, J.I. VEEDER and A.S. BAIN, "Large-scale Nuclear Energy from the Thorium Cycle", Proceedings of the 4th International Conference on the Peaceful Uses of Atomic Energy, 2, 239 (1971).

6. S. BANERJEE, E. CRITOPH and R.G. HART, "Thorium as a Nuclear Fuel for CANDU Reactors", Canadian Journal of Chemical Engineering, 53, 291 (1975).

7. E. CRITOPH, et al, "Prospects for Self-sufficient Equilibrium Thorium Cycles in CANDU Reactors", invited paper presented at the Winter Meeting of the American Nuclear Society, San Francisco, Nov. 16-21, 1975, Atomic Energy of Canada Limited, Report AECL-5501 (1976).

8. S.R. HATCHER, et al, "Thorium Cycle in Heavy Water Moderated Pressure Tube (CANDU) Reactors", Atomic Energy of Canada Limited, Report AECL-5398 (1976).

9. M.S. MILGRAM and W.H. WALKER, "Some Physics Problems in the Design of Thorium-fuelled CANDU Reactors", Atomic Energy of Canada Limited, Report AECL-5561 (1976).

10. E.C.W. PERRYMAN, "Nuclear Materials, Prospect and Retrospect", Atomic Energy of Canada Limited, Report AECL-5125 (1975).

11. R.W.F. GROSS, "Laser Isotope Separation", Optical Engineering, 13, (6), p. 506, (1974).

12. J.A.L. ROBERTSON, "Research and Development for Canadian Nuclear Power", Atomic Energy of Canada Limited, Report AECL-5314 (1976).

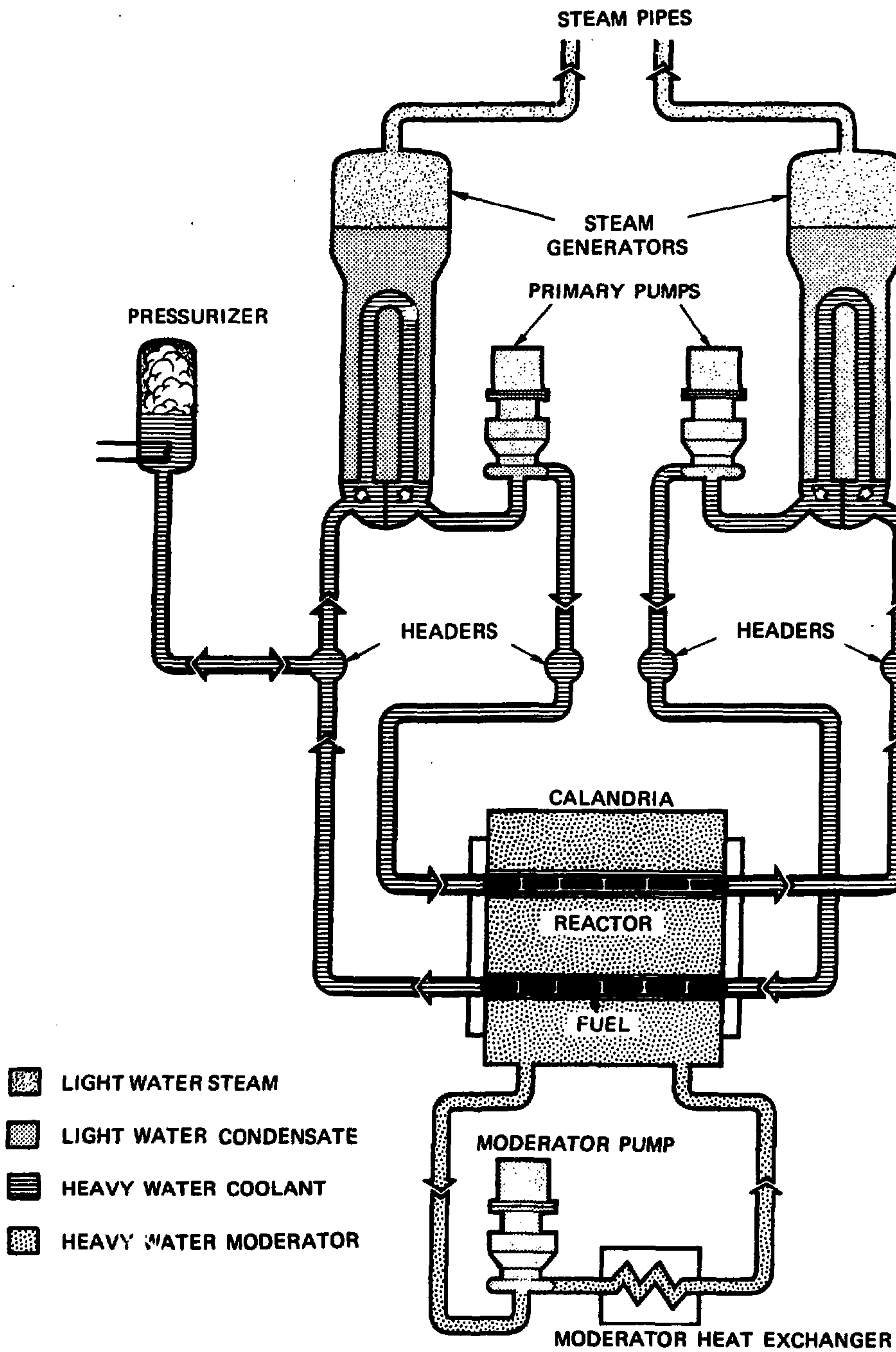
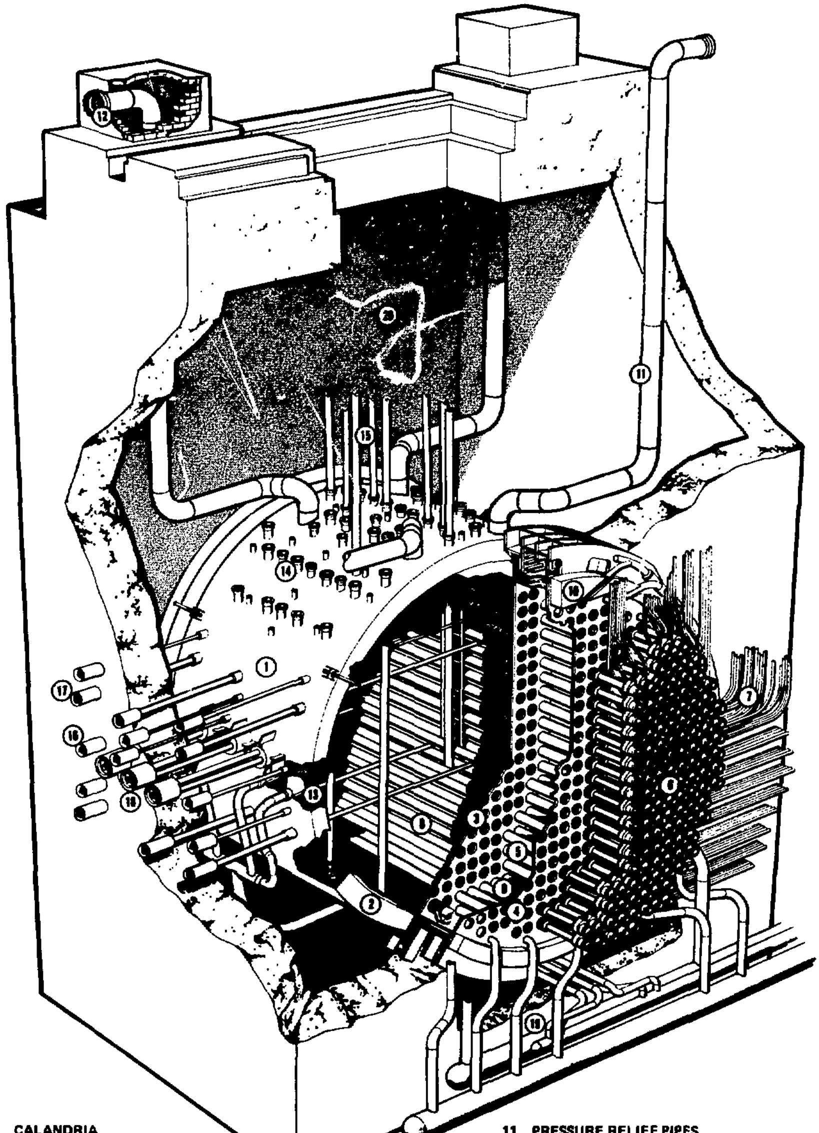
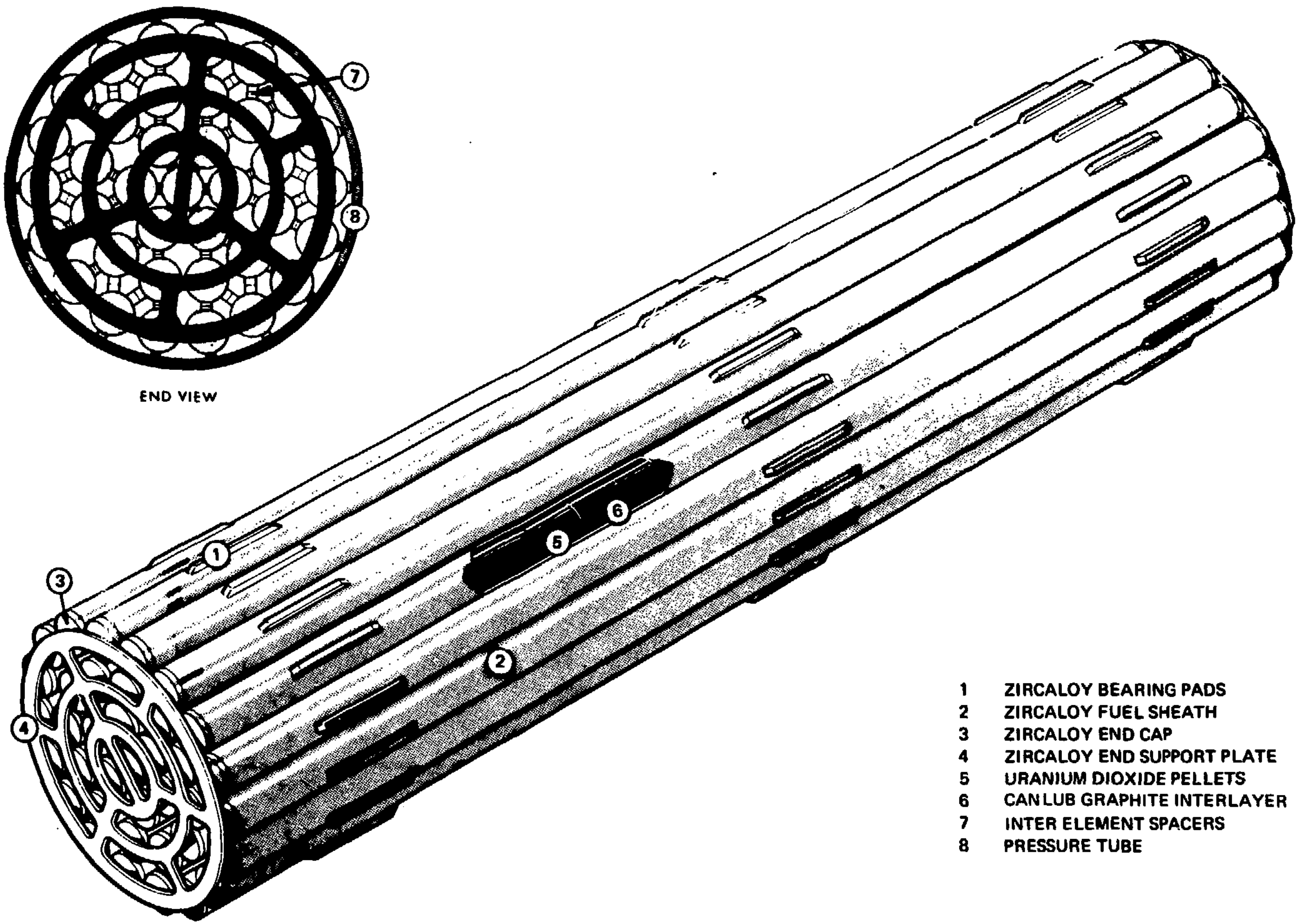


FIGURE 1 CANDU REACTOR SIMPLIFIED FLOW DIAGRAM



1 CALANDRIA
2 CALANDRIA SHELL
3 CALANDRIA SIDE TUBE SHEET
4 FUELLING MACHINE SIDE TUBE SHEET
5 LATTICE TUBES
6 *END FITTINGS*
7 FEEDERS
8 CALANDRIA TUBES
9 STEEL BALL SHIELDING
10 ANNULAR SHIELDING SLAB
11 PRESSURE RELIEF PIPES
12 RUPTURE DISC
13 MODERATOR INLETS (4 EACH SIDE)
14 REACTIVITY CONTROL NOZZLES
15 REACTIVITY CONTROL DEVICES
16 *HORIZONTAL FLUX DETECTORS (9)*
17 POISON INJECTOR NOZZLES (6)
18 *ION CHAMBER HOUSING (3 EACH SIDE)*
19 END SHIELD COOLING PIPING
20 CALANDRIA VAULT (LIGHT WATER SHIELD)

FIGURE 2 **600MW CANDU REACTOR**



1 ZIRCALOY BEARING PADS
2 ZIRCALOY FUEL SHEATH
3 ZIRCALOY END CAP
4 ZIRCALOY END SUPPORT PLATE
5 URANIUM DIOXIDE PELLETS
6 CANLUB GRAPHITE INTERLAYER
7 INTER ELEMENT SPACERS
8 PRESSURE TUBE

FIGURE 3 **37 ELEMENT FUEL BUNDLE**

59.37500-1
REV. 1 OCT. 197·
59 PSR

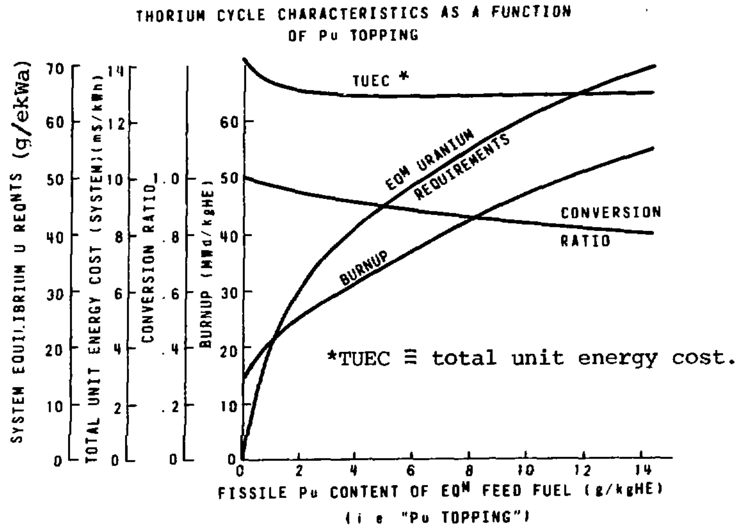


FIGURE 4

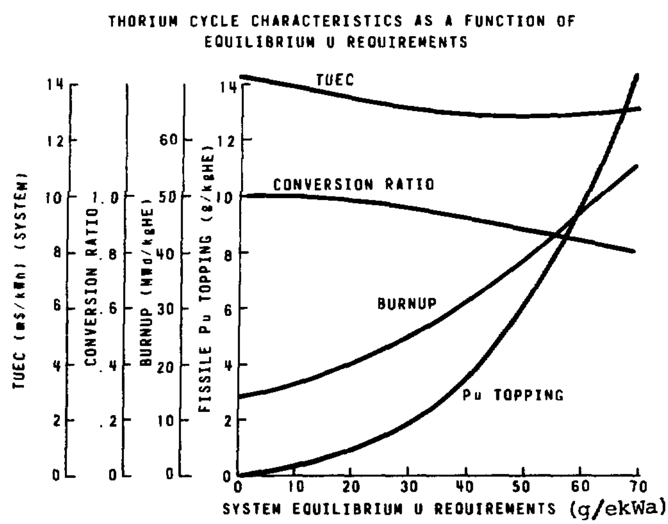


FIGURE 5

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