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Data Base for a CANDU-PHW Operating on a Once-Through, Slightly Enriched Uranium Cycle (AECL-6594)

AECL-6594

ATOMIC ENERGY
OF CANADA LIMITED

L'ÉNERGIE ATOMIQUE
DU CANADA LIMITÉE

DATA BASE FOR A CANDU-PHW OPERATING ON A ONCE-THROUGH, SLIGHTLY ENRICHED URANIUM CYCLE

Collection de données relatives à un réacteur CANDU-PHW fonctionnant avec un cycle de combustible à passe unique contenant de l'uranium légèrement enrichi

INFCE/WG 8/CAN/DOC 3

Chalk River Nuclear Laboratories

Laboratoires nucléaires de Chalk River

Chalk River, Ontario

July 1979 juillet

DATA BASE FOR A CANDU-PHW OPERATING ON A ONCE-THROUGH, SLIGHTLY ENRICHED URANIUM CYCLE

SUBMISSION

TO

INFCE WORKING GROUP 8

ADVANCED FUEL CYCLE AND REACTOR CONCEPTS

SUBGROUP A

ONCE-THROUGH FUEL CYCLES

ATOMIC ENERGY OF CANADA LIMITED
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Collection de données relatives à un réacteur CANDU-PHW fonctionnant avec un cycle de combustible à passe unique contenant de l'uranium légèrement enrichi

Résumé

Ce rapport, préparé pour INFCE* contient des données relatives à un concept extrapolé de réacteur CANDU-PHW de 1000 MWe fonctionnant avec un cycle de combustible à passe unique contenant de l'uranium légèrement enrichi, à savoir 1.2% en poids d'U-235. On commente les effets résultant des variations apportées à l'enrichissement du combustible, à la puissance maximale de canal et aux paramètres économiques.

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* INFCE: International Nuclear Fuel Cycle Evaluation

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DATA BASE FOR A CANDU-PHW OPERATING ON A ONCE-THROUGH, SLIGHTLY ENRICHED URANIUM CYCLE

ABSTRACT

This report, prepared for INFCE*, gives data for an extrapolated 1000 MW(e) CANDU-PHW design operating on a once-through fuel cycle with a feed fuel of slightly enriched uranium — 1.2 weight % U-235 in uranium. The effects of varying fuel enrichment, maximum channel power, and economic parameters are also discussed.

* International Nuclear Fuel Cycle Evaluation

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CONTENTS

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A ONCE-THROUGH, SLIGHTLY ENRICHED URANIUM CYCLE

INFCE/WG 8/CAN/DOC 3

1. *INTRODUCTION*

A fairly comprehensive description of a CANDU-PHW* operating on a once-through, natural uranium cycle was given in INFCE/WG 8/CAN/DOC 2. This was based on current Canadian designs and experience.

It is anticipated that essentially the same reactor system can be adapted to employ the once-through, slightly enriched uranium fuel cycle. Therefore, in this report, the above-mentioned document is assumed to be available as a basic reference and only differences associated with the change in fuel cycle will be discussed.

2. REFERENCE CASE

2.1 *Definition*

The reference case considered in this report is a 1000 MW(e) CANDU-PHW reactor unit in a 4-unit station. The feed fuel is assumed to be identical in design with that described in INFCE/WG 8/CAN/DOC 2, except that the uranium is 1.2 weight % U-235 in uranium rather than natural uranium.

2.2 *Major Facilities, Interconnections and Material Flows*

Figure 2.2 shows the major specialized facilities, their inter-relationship and the flow of strategic materials per 1000 MW(e) unit operating at equilibrium with an 80% load factor. The reactor inventories of heavy water and uranium are given on the figure. The delay times associated with each of the facilities are assumed to be as given in Table 2.2. It can further be assumed that no fuel need be supplied to the reactor (apart from the initial loading and six month reserve supply) until 275 full power days after initial start-up.

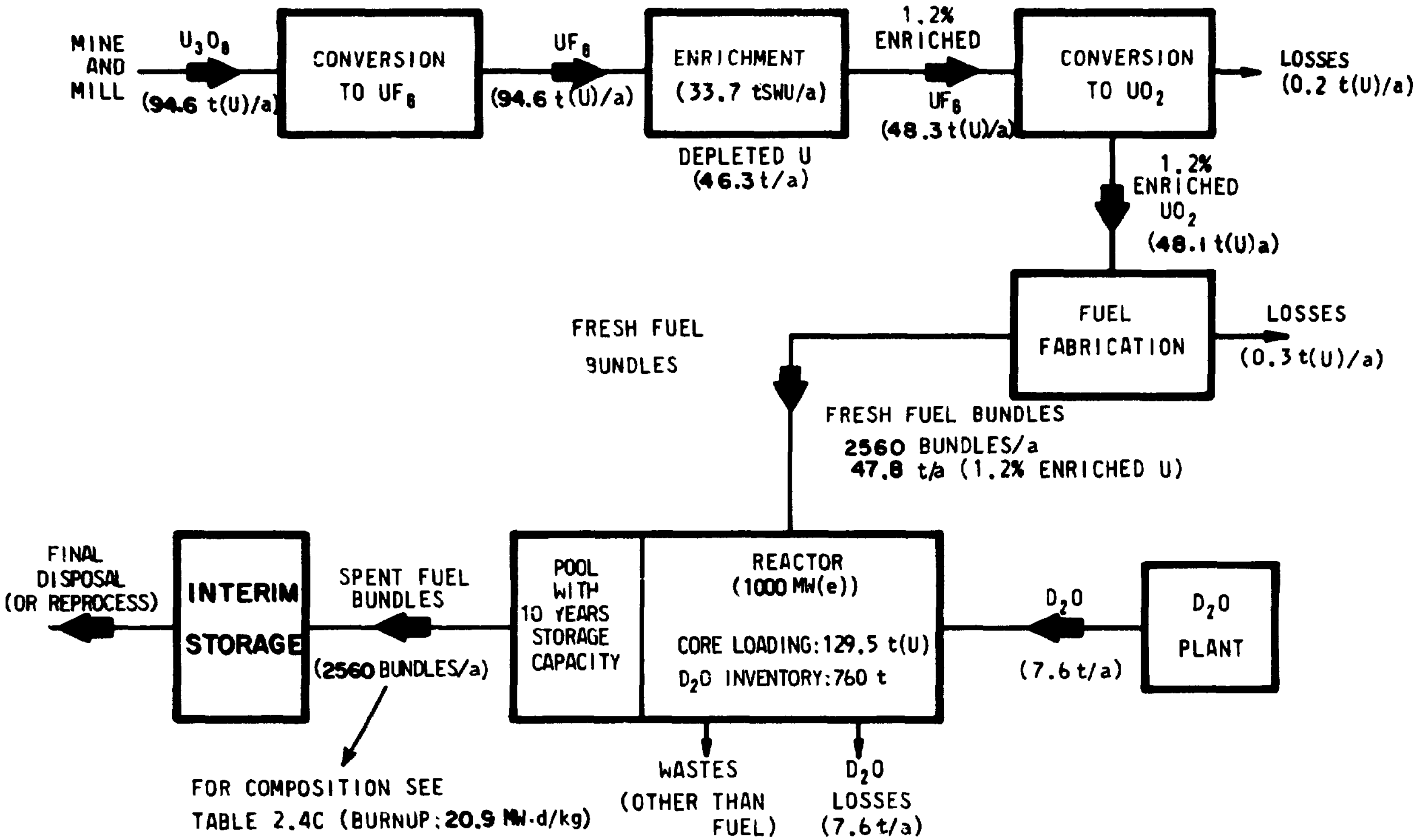
2.3 *System Design and Performance Data*

2.3.1 *Reactor System*

The reactor system design is assumed to be almost identical to a system of the same power designed for use with natural uranium. Minor changes

* CANDU-PHW signifies a CANadian design using Deuterium oxide (heavy water) as moderator, natural Uranium as fuel and Pressurized Heavy Water as coolant.

FIGURE 2.2
SCHEMATIC SHOWING MATERIAL FLOWS AT EQUILIBRIUM



ASSUMED LOAD FACTOR : 80%
ASSUMED ENRICHMENT TAILS : 0.2%

TABLE 2.2

DELAY TIMES FOR SLIGHTLY ENRICHED U FUEL CYCLE

Minehead to UF_6	:	9 months
UF_6 (natural) to UF_6 (enriched)	:	12 months
UF_6 (enriched) to fuel bundle	:	6 months

in design may be required in the new fuel storage and handling areas to accommodate the more reactive fresh fuel. Some modifications to the control and safety mechanisms may be required in order that their performance be preserved. In the case of the adjuster rods, the reactivity allowance has been adjusted to maintain the same decision and action time after a shutdown (before poisoning-out is caused by the xenon transient). However, these changes are judged to be relatively minor and have not yet been investigated thoroughly.

As will be discussed in the sections on "Technical Status and R, D & D Requirements" and "Effects of Variation of Parameters", there is a potential problem associated with an increase in power distribution *ripple* on refuelling.

The general reactor performance data are very similar to those for the once-through natural uranium cycle, and are summarized in Table 2.3.1.

The considerably longer burnup required of the fuel in the slightly enriched once-through cycle relative to the natural uranium one introduces some uncertainty as to whether the fuel design would need to be modified. Current judgment is that it would not, but considerable testing would be required to confirm this.

A fairly substantial number of bundles (∿5000) have been taken to average burnups in the range 10 000 - 15 000 MW·d/t(U). There have been no defects directly or indirectly attributable to high burnup alone. In the limited loop testing which has been done, there also have been no defects attributed to high burnup and, even when internal gas pressures were above coolant pressure, there were no unacceptable dimensional changes.

We do not have much information on the ability of the fuel design to accept a power ramp at burnups greater than 10 000 MW·d/t(U). A testing program to determine this type of behaviour could be carried out relatively easily, since there are more than 200 CANDU Pickering bundles with outer element burnups of 15 000 MW·t(U) or higher which could be used in the tests.

TABLE 2.3.1

GENERAL REACTOR PERFORMANCE SPECIFICATIONS

A. Power Plant Performance

Fission Power	:	3425 MW
Electrical Power, Gross	:	1074 MW
Net	:	1000 MW
Thermal Efficiency	:	29.2%

B. Reactor Parameters

Core Radius	:	3.871 m
Core Length	:	5.944 m
Core Volume	:	279.8 m^3 [2.798 x 10^5 L]

(N.B. A 0.655 m D_2O reflector surrounds core on sides]

Core Power Density	:	0.01224 MW/L
Coolant Flow Rate	:	11.7 Mg/s
Maximum Channel Flow	:	24 kg/s
Reactor Inlet Temperature	:	267°C
Reactor Outlet Temperature	:	310°C
Reactor Outlet Quality	:	4%
Reactor Outlet Pressure	:	10.0 MPa

C. Fuel Parameters

Maximum Fuel Temperature	:	1900°C
Maximum Cladding Surface Temp.	:	326°C
Initial Core Fuel Loading		
- total heavy metal	:	1.295 x 10^5 kg
- fissile material (U-235)	:	1554 kg
Discharge Exposure; average	:	20900 MW·d/t(U)
Conversion Ratio		
- beginning of life	:	0.57
- average, equilibrium cycle	:	0.70

2.3.2 *Fuel Fabrication*

2.3.2.1 *Process Description*

Powder Supply

Provision of low-enriched UO_2 feed to the pelletization, encapsulation and assembly plant requires three steps from receipt of uranium mill concentrate. These are:

(a) Conversion to UF_6 (natural)

(b) Enrichment, and

(c) Conversion of UF_6 to ceramic grade UO_2 (enriched).

These take the place of the single step in the natural CANDU cycle to produce natural UO_2 from the uranium mill concentrate.

The first conversion operation is already being carried out on a commercial scale in Canada during preparation of UF_6 for export. The process is well established, involving solvent extraction, denitration and reduction followed by successive reaction with HF and fluorine to form UF_6. The second conversion step could use water and ammonia reaction with the UF_6 to form ammonium diuranate (ADU) which is then hydrogen reduced to UO_2. This is called the wet-way route and is currently available in Canada only on a relatively small scale (2-3 Mg batches). A commercial sized operation would probably use a single step dry process involving steam and hydrogen reaction at about 600°C in a fluidized bed.

Pelletization, Encapsulation and Assembly

Manufacturing operations up to completed low-enriched CANDU bundles would be identical to those for natural fuel.

2.3 2.2 *Accountability*

If the plant were dedicated to the manufacture of a single enrichment, at least during campaigns of six months or more duration, then the added requirements for safeguards, security and accountability would be minor. If the plant were *flexible*, i.e., capable of producing fuel of various enrichments either simultaneously or in short campaigns, then the requirements for material accountability would be more substantial.

2.3.2.3 *Fuel Costs*

Conversion to UF_6

Conversion of uranium mill concentrate to natural UF_6 at a commercial throughput (several Mg/d) is estimated to cost 5$/kg U (1978 dollars are used throughout this report).

UF_6 to Completed Bundles

Fabrication costs from a dedicated or single enrichment plant using UF_6 feed would probably fall in the upper half of the 55 ±10$/kg HE range quoted for natural fuel. There would be a significant capital and operating cost penalty for a flexible or multi-enrichment plant as a result of the increased accountability requirements. The cost would then be in the range 70 ±10$/kg HE for the reference 37-element bundle design. These estimates assume private ownership of facilities, with appropriate commercial financing.

It is possible that fuel supply for several reactors at different stages in their approach to equilibrium will necessitate the flexible plant.

2.3.3 *Other Facilities*

Enrichment Plant

This cycle requires a facility for enriching uranium, although the separative work requirements per unit energy production are relatively small (see section 2.2). We have had no experience ourselves with enrichment facilities but these are being considered by INFCE WG-2.

Spent Fuel Storage Facilities

While the decay heat and radiation field as a function of time for spent fuel are higher than for the natural uranium case (see section 2.4) the differences are not expected to pose significant problems. In general, the spent fuel will be less reactive than spent fuel from the natural uranium cycle.

It is, therefore, expected that storage facilities would be very similar, if not identical, to those for spent natural uranium fuel. A somewhat longer cooling period before final disposal may be adopted.

Fuel Immobilization and Disposal

Again the procedures and facilities planned for handling spent natural uranium fuel should be adequate although, as mentioned above, somewhat longer cooling periods may be used before ultimate disposal of spent fuel.

2.4 *Fuel Management and Handling Information*

There are a number of plausible options for starting up a reactor on a once-through, slightly enriched uranium cycle. One would be to use a natural uranium initial loading, as outlined in INFCE/WG 8/CAN/DOC 2, but with 1.2% enriched uranium fuel as subsequent feed. Another would be to use 1.2% enriched uranium fuel for the initial loading as well as subsequent feed. In this option more reactivity hold-down would be required for the initial core. This could be in the form of boron in the moderator and/or use of some depleted uranium fuel bundles.

No detailed studies of the various start-up alternatives have been done. For conceptual simplicity and convenience in specifying data for strategic studies, it is assumed here that 1.2% enriched uranium fuel would be used throughout the reactor life, including the initial core loading. On the basis of experience with the natural uranium cycle, it is assumed that, with this scheme, no fuelling would be required for 275 full power days after start-up, at which time refuelling would start at the equilibrium rate. It is expected that uranium requirements based on these assumptions will be completely adequate for comparison studies.

Table 2.4 A summarizes some of the general fuel management information, and Table 2.4 B gives specific data in a form suitable for reactor strategy studies, such as those conducted by the IAEA. Note that the lifetime U_3O_8 requirements are only about 70% of those for the natural uranium once-through cycle as given in INFCE/WG 8/CAN/DOC 2.

Table 2.4 C gives the spent fuel isotopic composition (heavy elements) as a function of burnup.

The radiation field as a function of cooling time at a distance of one metre from a discharged bundle surface is shown in Figure 2.4 A. Figure 2.4 B shows the discharge fuel energy generation rate as a function of cooling time.

2.5 *Technical Status and R, D & D Requirements*

2.5.1 *Reactor Systems (General)*

The assumed reactor concept has been demonstrated in unit sizes up to 750 MW(e). The extensive development program underway for the 850 MW(e) and 1250 MW(e) reactors would also be applicable to a 1000 MW(e) design.

A few minor modifications might be required to adapt the design developed for operation on a natural uranium cycle, to operation on a slightly enriched uranium, once-through cycle. These would be in the areas of fresh fuel handling and storage, and in control and safety mechanisms. They would be expected to be straightforward.

Demonstration of the cycle would likely be achieved by full or partial conversion of an existing CANDU-PHW to slightly enriched uranium fuelling. It should be mentioned in this regard that the 25 MW(e) demonstration reactor, NPD, at Rolphton, Ontario has made extensive use of enriched (1.5%) uranium fuel for a number of years in connection with a cobalt-60 isotope production program. While this has provided valuable experience, it does not constitute a demonstration of the cycle, since the fuel ratings in NPD are considerably lower than those in commercial power reactors and burnups as high as 20 000 MW·d/t(U) have not been demanded.

2.5.2 *Thermohydraulic Performance*

Two potential problems of a thermohydraulic nature have been identified associated with changes in margin to dryout due to changes in power distributions.

a) The higher fuel neutron absorption cross-section for slightly enriched uranium, compared with natural uranium, results in a larger variation in bundle element powers. For a given bundle power, the maximum element power, which occurs in elements in the outer ring, is higher. Studies have shown that this problem can be solved by the use of enrichment grading, i.e., a higher initial fuel enrichment for elements in the inner rings than for those in the outer ring. By suitable enrichment grading, the ratio of maximum-to-average element power for a bundle can be kept well below that for a natural uranium bundle, throughout the bundle life. The fuel cycle properties are little affected by this grading, provided the average bundle enrichment is maintained.

TABLE 2.4 A

REACTOR FUEL MANAGEMENT INFORMATION

Average Capacity Factor	80%
Fraction of Core Replaced*	- On power refuelling is used with replacement of 8 (possibly 4 or 2 - see section 2.5.2) fuel bundles per channel visit. The nominal rate of fuelling is 7.0 bundles per average day (8.8 bundles per full power day). Therefore approximately one channel visit per day is required with 8/6924 of the core replaced at each visit.
Form of Fabricated Fuel	- 0.5 m long, 37 element bundles of Zr-clad UO_2 pellets with each bundle containing 18.7 kg of 1.2% enriched uranium.
Fuel Composition	- as loaded --- 1.2% enriched U - at discharge* --- see Table 2.4 C at 20 900 MW·d/t
Losses	- conversion --- 1/2% - fabrication -- 1/2%
U_3O_8 Requirements	- initial core : 256.0 t(U) - annual equilibrium : 94.6 t(U)/a - 30 year cumulative : 3005.0 t(U)
Separative Work Requirements	- initial core : 91.3 tSWU - annual equilibrium : 33.7 tSWU/a - 30 year cumulative : 1071.0 tSWU

* Note that refuelling does not start until after ~275 full power days operation and then proceeds at approximately the equilibrium rate. During the period between 275 full power days and 1000 full power days, the burnup of discharge fuel averages ~17 500 MW·d/t(U), and thereafter is at the equilibrium value of 20 900 MW·d/t(U). The average burnup of the fuel in the core increases over the first 1000 full power days to its equilibrium value of ~10 800 MW·d/t(U).

TABLE 2.4 B

FUEL CYCLE CHARACTERISTICS

CANDU-PHW OPERATING ON A SLIGHTLY ENRICHED, ONCE-THROUGH URANIUM FUEL CYCLE

Initial Loading		
	Uranium, t(U)/GW(e)	130
	Enrichment, % U-235	1.2%
	Heavy Water, $t(D_2O)/GW(e)$	760
Replacement Requirements		
a)	From start-up to 275 full power days (fpd)	
	Uranium, t(U)/(GW(e)·a)	0
	Heavy Water, $t(D_2O)/(GW(e)\cdot a)$	7.6
b)	From 275 fpd to 1000 fpd	
	Uranium, t(U)/(GW(e)·a)	59.8
	Enrichment, % U-235	1.2%
	Heavy Water, $t(D_2O)/(GW(e)\cdot a)$	7.6
c)	From 1000 fpd to end of life	
	Uranium, t(U)/(GW(e)·a)	59.8
	Enrichment, % U-235	1.2%
	Heavy Water, $t(D_2O)/(GW(e)\cdot a)$	7.6
Discharged Fuel		
a)	From start-up to 275 fpd	
	Uranium, t(U)/(GW(e)·a)	0
b)	From 275 fpd to 1000 fpd	
	Uranium, t(U)/(GW(e)·a)	58.4
	Enrichment, % U-235	0.16%
	Fissile Pu, $t(Pu_f)/(GW(e)\cdot a)$	0.20
	Total Pu, $t(Pu_t)/(GW(e)\cdot a)$	0.33
c)	From 1000 fpd to end of life	
	Uranium, t(U)/(GW(e)·a)	58.1
	Enrichment, % U-235	0.10%
	Fissile Pu, $t(Pu_f)/(GW(e)\cdot a)$	0.20
	Total Pu, $t(Pu_t)/(GW(e)\cdot a)$	0.36
Final Core		
	Uranium, t(U)/GW(e)	128
	Enrichment, % U-235	0.39%
	Fissile Pu, $t(Pu_f)/GW(e)$	0.39
	Total Pu, $t(Pu_t)/GW(e)$	0.56
	Heavy Water, $t(D_2O)/GW(e)$	760

Note: GW(e)·a is used here correctly to mean a full GWe for a full year. To account for load factor in determining annual uranium requirements and discharge rates merely multiply by the load factor. Heavy water replacement requirements are assumed to be independent of load factor.

TABLE 2.4 C

37-ELEMENT 1.2 WT % U235 ENRICHED FUEL

COOLANT DENSITY	0.80406.10**-6 GM/M**3	INITIAL WESTCOTT R-VALUE	0.05712
FUEL DENSITY	10.35900.10**-6 GM/M**3	INITIAL U-238 CAPTURE CROSS-SECTION	4.58970 BARNS*
FUEL TEMP	936.00000°C	EFFECTIVE NEUTRON TEMPERATURE	249.72177°C
EFFECTIVE CELL RADIUS	0.16122 M	INITIAL FAST FISSION RATIO	0.05665
INITIAL CONVERSION RATIO	0.57409		

------------------------------ U N I T S   A R E   G M / K G   (I N I T I A L   H. E.) ------------------------------

N/KB	MWD/T	U-235	U-236	NP-237	PU-238	PU-239	PU-240	PU-241	PU-242	TOTAL PU	CM-242	AM-241
0.0	0.	1.200E+01	0.	0.	0.	0.	0.	0.	0.	0.	0.	0.
.2	1314.	1.054E+01	2.311E-01	2.078E-03	2.897E-05	7.976E-01	3.539E-02	2.229E-03	5.514E-05	8.353E-01	6.084E-08	2.582E-06
.1	2628.	9.269E+00	4.319E-01	6.040E-03	1.868E-04	1.384E+00	1.288E-01	1.463E-02	7.900E-04	1.529E+00	1.869E-06	3.554E-05
.6	3921.	8.153E+00	6.060E-01	1.122E-02	5.291E-04	1.816E+00	2.563E-01	3.755E-02	3.267E-03	2.114E+00	1.173E-05	1.405E-04
.8	5184.	7.176E+00	7.568E-01	1.732E-02	1.086E-03	2.132E+00	4.033E-01	6.750E-02	8.299E-03	2.613E+00	3.915E-05	3.387E-04
1.0	6409.	6.321E+00	8.871E-01	2.405E-02	1.872E-03	2.362E+00	5.602E-01	1.016E-01	1.635E-02	3.042E+00	9.344E-05	6.298E-04
1.2	7594.	5.571E+00	9.996E-01	3.119E-02	2.886E-03	2.527E+00	7.201E-01	1.376E-01	2.761E-02	3.415E+00	1.817E-04	1.001E-03
1.4	8738.	4.914E+00	1.097E+00	3.855E-02	4.118E-03	2.644E+00	8.785E-01	1.743E-01	4.212E-02	3.743E+00	3.081E-04	1.434E-03
1.6	9841.	4.337E+00	1.180E+00	4.600E-02	5.552E-03	2.726E+00	1.032E+00	2.106E-01	5.978E-02	4.034E+00	4.743E-04	1.911E-03
1.8	10906.	3.831E+00	1.252E+00	5.341E-02	7.167E-03	2.781E+00	1.180E+00	2.458E-01	8.045E-02	4.294E+00	6.796E-04	2.413E-03
2.0	11936.	3.386E+00	1.314E+00	6.070E-02	8.942E-03	2.816E+00	1.319E+00	2.795E-01	1.039E-01	4.528E+00	9.218E-04	2.926E-03
2.2	12932.	2.995E+00	1.367E+00	6.783E-02	1.085E-02	2.838E+00	1.450E+00	3.115E-01	1.299E-01	4.740E+00	1.198E-03	3.437E-03
2.4	13897.	2.651E+00	1.413E+00	7.474E-02	1.287E-02	2.849E+00	1.573E+00	3.415E-01	1.583E-01	4.935E+00	1.503E-03	3.937E-03
2.6	14835.	2.347E+00	1.452E+00	8.140E-02	1.499E-02	2.853E+00	1.688E+00	3.696E-01	1.888E-01	5.114E+00	1.834E-03	4.421E-03
2.8	15748.	2.080E+00	1.485E+00	8.781E-02	1.718E-02	2.852E+00	1.794E+00	3.957E-01	2.212E-01	5.280E+00	2.186E-03	4.882E-03
3.0	16639.	1.844E+00	1.513E+00	9.395E-02	1.942E-02	2.847E+00	1.893E+00	4.200E-01	2.553E-01	5.435E+00	2.555E-03	5.319E-03
3.2	17509.	1.636E+00	1.537E+00	9.983E-02	2.169E-02	2.840E+00	1.984E+00	4.425E-01	2.909E-01	5.579E+00	2.937E-03	5.730E-03
3.4	18361.	1.452E+00	1.557E+00	1.054E-01	2.399E-02	2.832E+00	2.069E+00	4.633E-01	3.278E-01	5.715E+00	3.328E-03	6.115E-03
3.6	19197.	1.290E+00	1.573E+00	1.108E-01	2.630E-02	2.822E+00	2.147E+00	4.826E-01	3.659E-01	5.844E+00	3.724E-03	6.474E-03
3.8	20019.	1.146E+00	1.587E+00	1.158E-01	2.861E-02	2.813E+00	2.219E+00	5.003E-01	4.049E-01	5.965E+00	4.124E-03	6.809E-03
4.0	20828.	1.019E+00	1.598E+00	1.207E-01	3.091E-02	2.803E+00	2.285E+00	5.168E-01	4.448E-01	6.081E+00	4.523E-03	7.119E-03
4.2	21626.	9.066E-01	1.607E+00	1.253E-01	3.319E-02	2.794E+00	2.346E+00	5.319E-01	4.854E-01	6.190E+00	4.921E-03	7.407E-03
4.4	22414.	8.070E-01	1.613E+00	1.296E-01	3.544E-02	2.785E+00	2.402E+00	5.459E-01	5.265E-01	6.295E+00	5.314E-03	7.675E-03
4.6	23193.	7.186E-01	1.618E+00	1.338E-01	3.766E-02	2.777E+00	2.454E+00	5.588E-01	5.682E-01	6.396E+00	5.702E-03	7.923E-03
4.8	23965.	6.403E-01	1.622E+00	1.377E-01	3.985E-02	2.769E+00	2.502E+00	5.707E-01	6.103E-01	6.492E+00	6.082E-03	8.154E-03
5.0	24730.	5.708E-01	1.624E+00	1.414E-01	4.199E-02	2.761E+00	2.546E+00	5.817E-01	6.526E-01	6.584E+00	6.454E-03	8.369E-03
5.2	25489.	5.090E-01	1.625E+00	1.450E-01	4.409E-02	2.754E+00	2.587E+00	5.919E-01	6.952E-01	6.673E+00	6.817E-03	8.568E-03
5.4	26243.	4.542E-01	1.625E+00	1.483E-01	4.614E-02	2.748E+00	2.625E+00	6.013E-01	7.379E-01	6.758E+00	7.170E-03	8.754E-03
5.6	26992.	4.055E-01	1.624E+00	1.515E-01	4.815E-02	2.742E+00	2.659E+00	6.101E-01	7.807E-01	6.840E+00	7.511E-03	8.928E-03
5.8	27738.	3.622E-01	1.622E+00	1.545E-01	5.010E-02	2.737E+00	2.691E+00	6.181E-01	8.235E-01	6.920E+00	7.842E-03	9.090E-03
6.0	28480.	3.236E-01	1.620E+00	1.573E-01	5.200E-02	2.731E+00	2.721E+00	6.256E-01	8.663E-01	6.996E+00	8.161E-03	9.241E-03

* 1 b = 10^{-28} m^2

FIGURE 2.4 A

FIELD AS A FUNCTION OF COOLING TIME FOR A
1.2% ENRICHED IRRADIATED CANDU BUNDLE
(BURNUP: 20,000 MW·d/t(U))

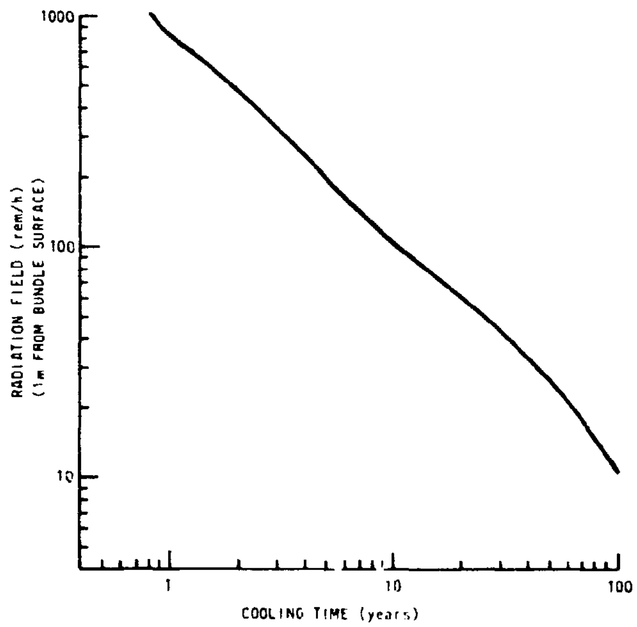
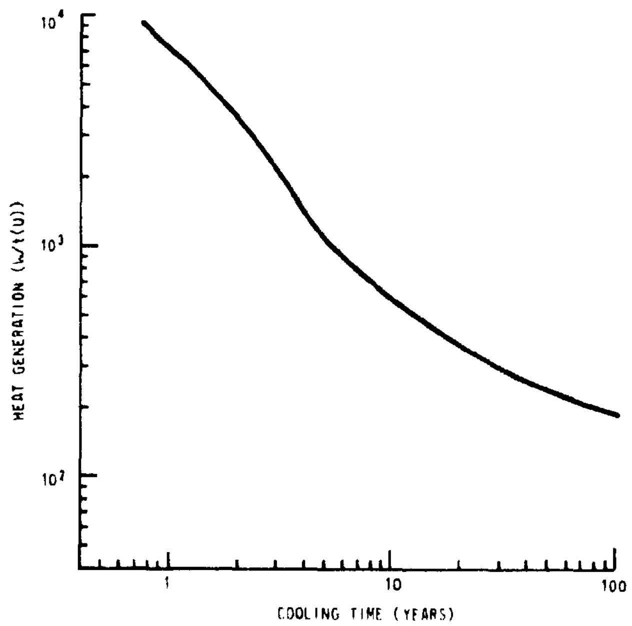


FIGURE 2.4 B

DISCHARGE FUEL ENERGY GENERATION RATE AS
A FUNCTION OF COOLING TIME
(1.2% ENRICHED FUEL, BURNUP: 20,000 MW·d/t (U))



b) The second potential problem is associated with the power distribution *ripple* produced by continuous, on-line refuelling. The variation, as a function of energy output, on both reactivity and fission cross-section in slightly enriched uranium fuel is greater than for natural uranium fuel. Thus the local increase in neutron flux and power, at constant reactor power, associated witn a given on-line refuelling event is larger. Other things being equal, this would raise the design maximum channel power.

This type of effect has been studied to a certain extent, but unfortunately not for this particular reference design. As indicated below, there are a number of possible solutions to the potential problem.

i) Fuel Management:

Since the discharge burnup is much higher for slightly enriched fuel than natural fuel, a four (or, possibly, two) bundle shift scheme could be considered rather than the reference eight bundle shift, without straining the capacity of the fuelling machines. This would substantially reduce the *ripple* effect but a residue, higher than the natural uranium case, would remain. This would also impose more rigorous requirements on fuel performance with respect to power increases.

ii) Zonal Control:

A stronger, more distributed zonal control system could be provided. This is not thought to be a practical way of completely resolving the problem, but may be useful as a partial solution.

iii) Fuel Design:

There is scope for relatively small changes in fuel design to at least partially improve the ability to accept somewhat higher maximum channel powers. The enrichment grading mentioned above might provide some flexibility in this respect. Improvements already made (e.g., CANLUB) may permit use of fuel management schemes mentioned under i) above without signigicant risk of fuel defects.

iv) Reactor De-rating:

If none of the above solutions separately or in combination are satisfactory then existing reactors might have to be de-rated a few per cent if converted to slightly enriched fuelling, and new reactors may have to be designed with a few additional channels for a given reactor power.

In any event, it is clear that additional research and development associated with this potential problem should precede large-scale introduction of the slightly enriched, once-through uranium cycle.

2.5.3 *Fuel Development*

On the basis of the limited testing of fuel to extended burnup which has been done, it is expected that the current fuel design would be adequate.

Nevertheless, further irradiation testing of fuel to high burnup would be required to achieve sufficient confidence for widespread introduction of the cycle. The possibility of a requirement for larger fission product gas plenums does exist.

Irradiation testing would also be necessary for graded enrichment designs.

2.5.4 *Enrichment Capability*

A very large R, D & D program would be required to provide independent enrichment facilities in Canada. The alternative of buying enrichment services abroad would, of course, require no development.

2.6 *Safety and Accident Considerations*

While no detailed design investigation has been carried out, no significant difficulties in meeting the same criteria as those for the natural uranium case are foreseen.

The major difference is probably the increased fission product hold-up in the reactor core. Thus fission product releases associated with a given number of fuel defects would be larger.

2.7 *Environmental Information*

The environmental effects of the reactor system would be similar to those given in INFCE/WG 8/CAN/DOC 2. Fission product releases from a fuel defect would be expected to be larger but given the same defect rate (in terms of fraction of elements defecting) the overall release would be similar.

Of course, environmental effects associated with mining and milling would be reduced due to improved uranium utilization. At least partially balancing this improvement would be additional effects due to the increased processing required (e.g., enrichment). Fuel disposal masses would be reduced, inversely proportional to fuel burnup.

2.8 *Non-Proliferation Data*

2.8.1 *Safeguards*

Reactor safeguards similar to those discussed in INFCE/WG 8/CAN/DOC 2 would be equally applicable and effective.

2.8.2 *Enrichment Facility*

The significant new feature in the cycle affecting non-proliferation considerations is the enrichment facility. This would presumably require appropriate safeguards as will, no doubt, be discussed by Working Group 2. If the alternative of buying enrichment services abroad were adopted, no significant increase in proliferation risk would be involved other than a marginally more sensitive material as feed fuel.

2.8.3 *Non-Proliferation Information*

The data on spent fuel accumulation rates have already been given, as has the isotopic composition of the spent fuel and radiation fields associated with it.

Some comments are given below on elements of assessment of proliferation resistance that have been suggested.

Safeguardability

An acceptable safeguards scheme has been devised for irradiated fuel on the reactor site. This type of scheme lends itself to further extension to cover interim storage or ultimate disposal sites.

Weapons Usability

The fresh fuel in this cycle is 1.2% enriched uranium and hence not directly usable for nuclear weapons. The spent fuel contains about 6 g of plutonium (total) per kg of heavy metal. If chemically separated from the heavy metal this would be usable for nuclear weapons.

Material Modifiability

As mentioned above, plutonium could be obtained by chemical separation from the spent fuel. The content of plutonium in the spent fuel is relatively low ($\sim$6 g/kg HE), the isotopic composition is not ideal (>35% Pu-240), and the fuel is initially highly radioactive with a relatively slow decay (see Figure 2.4 A). All these factors would make the separation of plutonium for weapons use relatively unattractive.

Facility Modifiability

The process could be altered to produce a more favourable plutonium composition for nuclear weapons by substantially reducing the burnup of the discharge fuel. This should be easily detected. There is little scope for increasing the concentration of plutonium in the spent fuel.

Interruptability

The inspection frequency in the safeguards scheme is such that diversions would be detected relatively promptly.

Material Accessibility

Spent fuel has a low vulnerability to theft or seizure, mainly due to the relatively low plutonium concentration ($\sim$6 g/kg HE) and the relatively high radiation fields.

2.9 *Economics*

2.9.1 *Introduction*

The base case used for economic studies is a conceptual 4 x 1000 MW(e) CANDU-PHW station, using 1.2 weight % U-235 enriched UO_2 fuel, identical in all respects (other than those directly attributable to the use of enrichment) to that described in INFCE/WG 8/CAN/DOC 2. Table 2.9.1 lists the basic parameters of the reference 1.2% enriched uranium once-through case as well as the natural uranium once-through reference case from INFCE/WG 8/CAN/DOC 2.

As for the natural uranium cycles, an up-dated version of the computer code, CANCAP-1973 (Ref. 1), was used to provide estimated costs. The same code was used to study the variation of important parameters as discussed in section 3.

2.9.2 *Ground Rules*

The ground rules adopted were the same as those used in INFCE/WG 8/CAN/DOC 2 (Section 2.8.2).

The effects of using a higher burnup fuel on replacement fuel purchase date and commencement of refuelling date are shown in Table 2.9.2 A.

The reference values used for the main relevant economic parameters are given in Table 2.9.2 B.

2.9.3 *Costing Method*

The costing method used was identical to that of INFCE/WG 8/CAN/DOC 2 (section 2.8.3).

2.9.4 *Capital Costs*

Table 2.9.4 compares data for 4 x 1000 MW(e) stations using natural uranium and 1.2% U-235 enriched uranium fuel. Total capital costs are ∿4% higher for the enriched reactor – the difference being largely attributable to the higher costs of the initial fuel inventory.

2.9.5 *Discounted Capitalized Operational Costs*

Table 2.9.5 compares data for corresponding natural and 1.2% enriched reactors. Total discounted operational costs are lower by ∿12% for the enriched reactor, largely due to a lower *replacement fuel* cost.

2.9.6 *Discounted Unit Cost and Product Cost*

In Table 2.9.6 the capital and operational costs have been expressed as unit costs (i.e., dollars/kW(e)), and as average product cost. The enriched reactor has a product cost ∿3% lower than that for the equivalent natural uranium burning reactor.

TABLE 2.9.1

BASIC PARAMETERS OF REFERENCE REACTORS

Paramaters	4 x 1000 MW(e) CANDU-PHW	4 x 1000 MW(e) 1.2% U-235 CANDU-PHW
Core Radius (m)	3.871	→
Core Length (m)	5.944	→
Reflector Thickness (m)	0.655	→
Lattice Pitch (m)	0.28575	→
No. of Channels	577	→
Pressure Tube ID (cm)	10.34	→
Heavy Water Purity (wt% D_2O)	99.75	→
Radial Form Factor	0.85	→
Max. Channel Power (MW)	6.6	→
Reactivity Load (mk)	18.5	15.5
Fuel Enrichment	Natural	1.2 wt% U-235
Elements per Fuel Bundle	37	→
Average Fuel Burnup (MW·h/kg(U))	176	502
Average Fuel Burnup (MW·d/t(U))	7300	20 900
No. of Coolant Loops	3	→
No. of Pumps per Loop	2	→
No. of Boilers per Loop	2	→
Heat Transfer Area/Boiler (m^2)	3320	→
Channel Outlet Press (MPa)	10.29	→
Channel Outlet Temp. (°C)	312	→
Channel Outlet Quality (%)	2.9	→
Net Electric (MW(e))	1000	→
Net Station Efficiency (%)	29.2	→
Main Heat Rejection	Fresh Water	→

TABLE 2.9.2 A

KEY DATES* FOR REACTOR CONSTRUCTION AND OPERATION

1.2% U-235 ENRICHED, 1000 MW(e)

	4 x 1000 MW(e) CANDU-PHW	4 x 1000 MW(e) 1.2% U-235 CANDU-PHW
Reference Date	1978.0	1978.0
Start of Construction	1978.0	1978.0
Heavy Water Purchase	1983.5	1983.5
Initial Fuel Charge Purchase	1983.5	1983.5
Reactor in Service	1984.0	1984.0
Annual O & M and Heavy Water Upkeep Begins	1984.5	1984.5
Commence Purchase Replacement Fuel	1984.0	1984.5
Commence Refuelling	1984.5	1985.0
Reactor Shutdown, Decommissioning	2014.0	2014.0

* Terminology Note: 1978.0 means beginning January 1978, 1984.5 means end June 1984, etc.

TABLE 2.9.2 B

BASE VALUES OF ECONOMIC PARAMETERS* FOR REFERENCE

1.2% U-235 ENRICHED, 1000 MW(e) REACTOR

U_3O_8 Cost	99 \$/kg (117 \$/kg(U))
Heavy Water Cost	270 \$/kg
Separative Work Cost	100 \$/SWU
Enrichment Plant Tails	0.2% U-235
Fuel Fabrication Cost	70 \$/kg(U)[a]
Fabricated Fuel Bundle Cost	405 \$/kg(U)[b]
Plant Capacity Factor	80%
Plant Life	30 years
Heavy Water Upkeep Rate	0.6 kg·h/primary circuit loop
Shipping and Repository Fuel Disposal Cost	49 \$/kg(U)[c]
Escalation Rate	0%
Interest Rate	4%
Discount Rate	4%
Cost of Decommissioning Reactor at End of Life	42 M\$

* All costs are in 1978 Canadian dollars
a) Includes no carrying charges.
b) Includes 23 \$/kg(U) carrying charges.
c) Based on scaling the cost estimates for natural uranium fuel with burnup, and is probably too high relative to disposal of natural uranium spent fuel.

TABLE 2.9.4

CANDU-PHW ONCE-THROUGH DIRECT AND INDIRECT CAPITAL COST ESTIMATES

Description	4 x 1000 MW(e) Natural Ref. Case (M$) (per unit)	4 x 1000 MW(e) 1.2% U-235 Ref. Case (M$) (per unit)
Site/Improvements	0.6	0.6
Buildings/Structures	84.3	82.3
Reactor, Boiler, Auxiliaries	114.8	114.8
Turbine, Gen. and Aux.	76.7	76.7
Electric Power Systems	36.8	36.8
Control and Instruments	15.1	15.1
Common Processes & Services	22.4	22.4
Total Direct Capital Cost	350.7	348.7
Engineering & Other Services	145.2	152.9
Heavy Water Inventory	210.0	210.0
Contingencies	45.8	47.0
Initial Fuel Inventory	22.6	53.2
Commissioning Cost	33.1	33.1
Interest to Inservice Date	72.6	74.0
Total Indirect Capital Cost	529.3	570.2
Total Capital Cost (incl. fuel)	879.9	918.9

TABLE 2.9.5

DISCOUNTED OPERATIONAL COSTS

Description	4 x 1000 MW(e) Natural Ref. Case (M$) (per unit)	4 x 1000 MW(e) 1.2% U-235 Ref. Case (M$) (per unit)
O and M	143.6	143.2
Heavy Water Upkeep	75.0	75.0
Replacement Fuel	421.5	335.5
Spent Fuel Disposal	29.5	31.1*
Decommissioning	13.0	13.0
Total Discounted Capitalized Operational Costs	682.6	597.9

* See footnote c) on Table 2.9.2 B

TABLE 2.9.6

UNIT COST AND PRODUCT COST SUMMARY

Description	4 x 1000 MW(e) Natural Ref. Case	4 x 1000 MW(e) 1.2% U-235 Ref. Case
Unit Initial Capital ($/kW(e))	879.9	918.9
Unit Total Discounted Capitalized Operating Cost ($/kW(e))	682.6	597.9
Total Discounted Unit Cost ($/kW(e))	1562.5	1516.8
Average Product Cost (M$/(kW·h))	12.9	12.5

2.9.7 *Variation in Economic Parameters*

Variation in interest and discount rates, heavy water price, U_3O_8 price, SWU cost and fuel fabrication cost were studied. The results are shown in Figures 2.9.7 A to 2.9.7 C for a range of interest rates from 0 to 12%/a and for discount minus interest rates from 0 to 12%/a. Table 2.9.7 A shows the sensitivity of unit and product costs to heavy water price over the range from 150 to 450 (1978 Canadian $)/kg. Table 2.9.7 B shows the effects of a range of U_3O_8 prices from 99 to 220 (1978 Canadian $)/kg. Table 2.9.7 C illustrates the effect on costs of a range of SWU costs from 50 to 150 (1978 Canadian $)/SWU, and Table 2.9.7 D shows the effects of fuel fabrication costs over the range 55 to 85 (1978 Canadian $)/kg U.

2.9.8 *References for Section 2.9*

(1) J.R. Frey et al, CANCAP 1973, Atomic Energy of Canada Limited unpublished report, TDVI-334.

3. EFFECTS OF VARIATION OF PARAMETERS

3.1 *Fuel Enrichment*

Table 3.1 lists fuel utilization and economic data for the reference 1000 MW(e) reactor using fuel of different enrichments. The increase in total capital cost with enrichment (due to initial fuel inventory costs) is compensated to varying degrees by better fuel utilization achieved.

There is a broad optimum in terms of average product cost as a function of fuel enrichment. Over this optimum the average product cost is about 3% lower than that for the natural uranium case.

3.2 *Reduction in Maximum Channel Power*

Table 3.2 shows the result of decreasing maximum channel power by 10% while increasing the number of channels by 10%. Although there is $\sim$1% increase in burnup, total capital costs increase by $\sim$3% and average product cost by $\sim$1%.

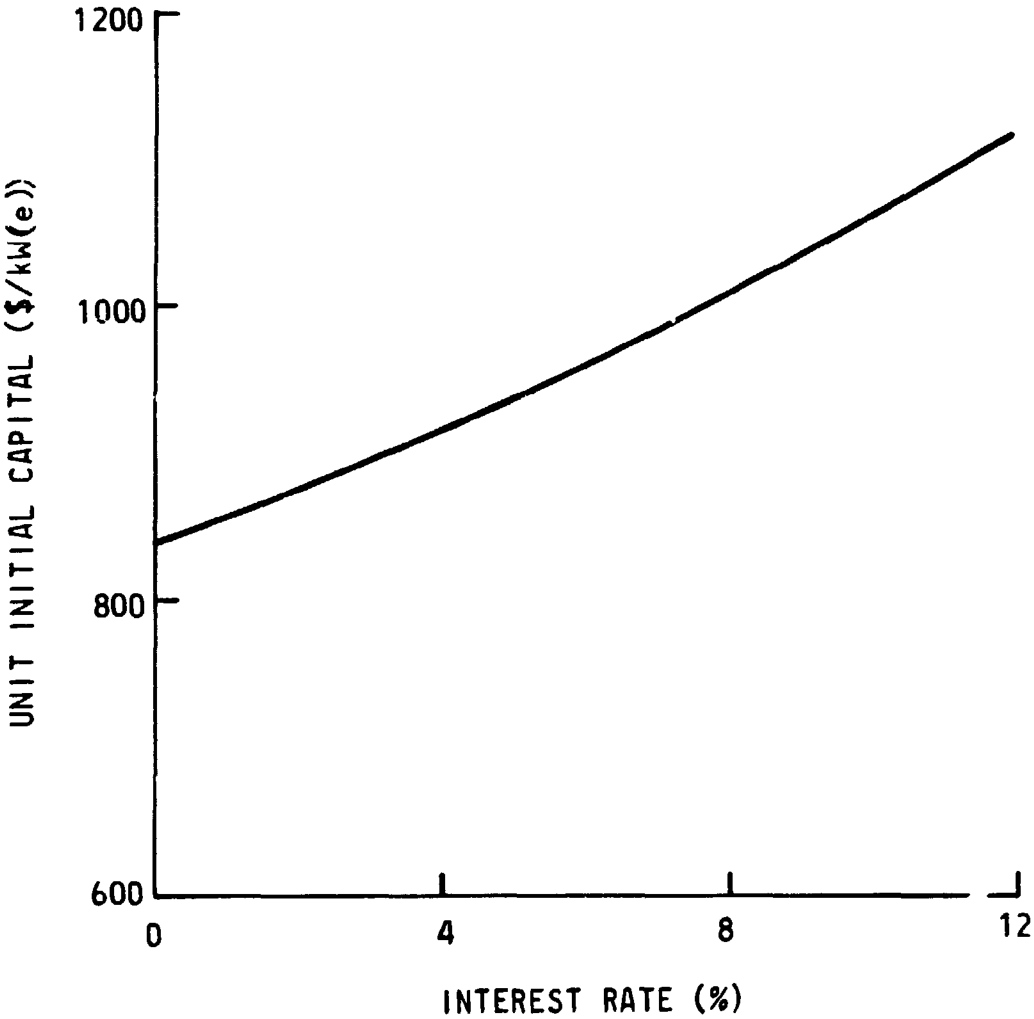


FIGURE 2.9.7 A

UNIT CAPITAL COST vs INTEREST RATE

(1.2% ENR. U, 4 x 1000 MW(e) STATION)

FIGURE 2.9.7 B

UNIT TOTAL DISCOUNTED CAPITALIZED OPERATING COST

VS

DISCOUNT RATE

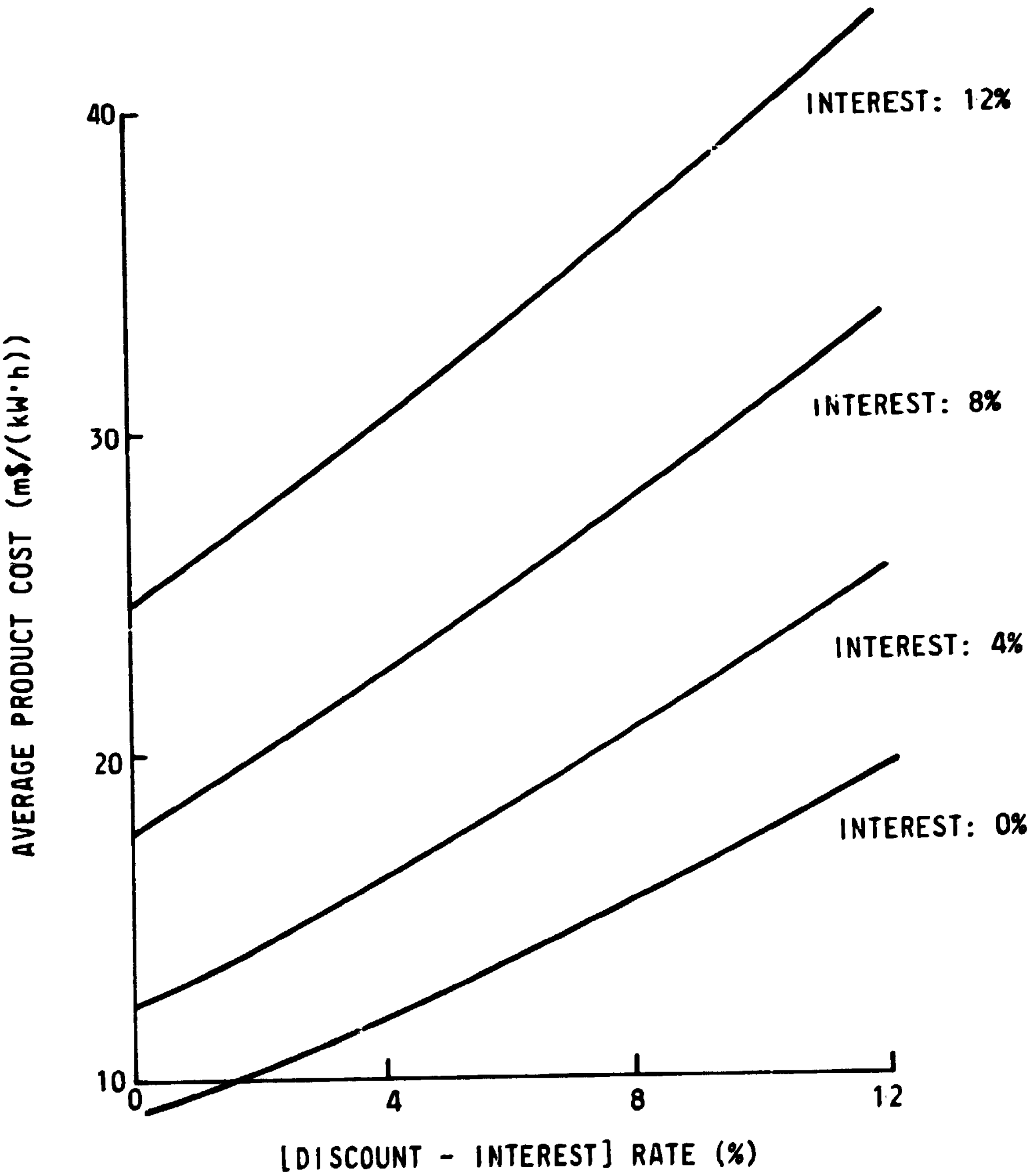


FIGURE 2.9.7 C

AVERAGE PRODUCT COST vs (DISCOUNT - INTEREST) RATE FOR VARIOUS INTEREST RATES. (1.2% ENR. U, 4 x 1000 MW(e) STATION)

TABLE 2.9.7 A

STATION COSTS vs HEAVY WATER COST

1.2% U-235 Enriched, 1000 MW(e)

	Heavy Water Cost ($/kg)		
	150	270	450
Unit Initial Capital ($/kW(e)	820.3	918.9	1066.9
Total Discounted Unit Cost ($/kW(e))	1384.7	1516.8	1714.9
Average Product Cost (M$/kW·(h))	11.4	12.5	14.1

TABLE 2.9.7 B

STATION COSTS vs U_3O_8 COST

1.2% U-235 Enriched, 1000 MW(e)

	U_3O_8 Cost ($/lb)/($/kg)			
	45/99	60/132	80/176	100/220
Fuel Bundle Cost ($/kg(U))	405	489	601	714
Total Discounted Fuel Cost (M$/(kW·h))	3.2	3.9	4.8	5.7
Average Product Cost (M$/(kW·h))	12.5	13.2	14.1	15.1

TABLE 2.9.7 C

STATION COSTS vs SWU COST

1.2% U-235 Enriched, 1000 MW(e)

	SWU COST ($/SWU)		
	50	100	150
Fuel Bundle Cost ($/kg(U))	369	405	441
Total Discounted Fuel Cost (M$/(kW·h))	2.9	3.2	3.5
Average Product Cost (M$/(kW·h))	12.2	12.5	12.8

TABLE 2.9.7 D

STATION COSTS vs FUEL FABRICATION COST

1.2% U-235 Enriched, 1000 MW(e)

	Fuel Fabrication Cost ($/kg(U))		
	55	70	85
Fuel Bundle Cost ($/kg(U))	390	405	420
Total Discounted Fuel Cost (M$/(kW·h))	3.1	3.2	3.3
Average Product Cost (M$/(kW·h))	12.4	12.5	12.6

TABLE 3.1

ECONOMIC DATA AS FUNCTION OF FUEL ENRICHMENT FOR 4 x 1000 MW(e) UNIT

Description	Natural UO_2	0.9% U-235 Enriched	1.2% U-235 Enriched	1.4% U-235 Enriched
Burnup (MW·h/kg(U))	176	334	502	595
Burnup (MW·d/t(U))	7300	13 900	20 900	24 800
Fuel Bundle Cost ($/kg(U))	172	279	405	493
Total Direct Capital Cost (M$)	350.7	349.2	348.7	348.5
Initial Fuel Inventory (M$)	22.6	36.7	53.2	64.8
Total Capital Cost Including Fuel (M$)	879.9	896.9	918.9	934.6
Discounted Replacement Fuel Cost (M$)	421.5	351.4	335.5	342.0
Total Discounted Capitalized Operational Costs (M$)	682.6	612.7	597.9	604.7
Total Discounted Unit Cost ($/kW(e))	1562.5	1509.6	1516.8	1539.2
Discounted Fuel Cost (M$/(kW·h))	3.56	3.21	3.22	3.36
Average Product Cost (M$/kW·h))	12.9	12.5	12.5	12.7

TABLE 3.2

EFFECT OF DOWNRATING CHANNEL POWER

Description		4 x 1000 MW(e) Natural Ref. Case	4 x 1000 MW(e) 1.2% U-235 Ref. Case	4 x 1000 MW(e) 1.2% U-235 Downrated 10%
Burnup	(MW·h/kg(U))	176	502	507
	(MW·d/t(U))	7 300	20 900	21 100
Maximum Channel Power (MW)		6.6	6.6	5.94
Number of Channels		577	577	641
Unit Initial Capital ($/kW(e))		879.9	918.9	944.1
Unit Total Discounted Capitalized Operation Cost ($/kW(e))		682.6	597.9	594.4
Total Discounted Unit Cost ($/kW(e))		1562.5	1516.8	1538.5
Average Product Cost (M$/(kW·h))		12.9	12.5	12.7

APPENDIX A

ABBREVIATIONS USED ON COMPUTER LISTINGS IN THIS REPORT

Computer Abbreviation	Accepted Abbreviation	Definition
CEL	C	Celsius
CM	cm	centimetre
KG	kg	kilogram
KG/S	kg/s	kilogram per second
KW	kW	kilowatt
KW/M	kW/m	kilowatt per metre
MEV	MeV	million electron volt
MPA	MPa	megapascal
MW	MW	megawatt
MWD	MW·d	megawatt day
MWH/KG	MW·h/kg	megawatt hour per kilogram
N/CM2-S	$n/(cm^2 \cdot s)$	neutron per centimetre squared second
N/KBARN	n/kbarn (1 b = 100 fm^2)	neutron per kilobarn (100 square femtometres)
Q/4PI	$Q/4\pi$	heat flux/4π

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