REALISTIC EVALUATION OF THE NATURAL URANIUM DEMAND OF NUCLEAR POWER SYSTEMS

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1. Introduction

As pessimistic predictions concerning future power supply have been predominating, recently the fuel cycle of nuclear power has been of increasing concern all over the world. The keen interest can be no doubt attributed to the fact that man faces difficulties in fuel supply also in the field of nuclear power production in spite of the estimate according to which fission based nuclear power is the only viable approach to cope with the world-wide energy crisis in the long run. A remedy to this problem may be the wide spread of nuclear power plants operating with fast breeder reactors anticipated by the end of this century. By this time, however, nuclear power plants operating with thermal reactors continue predominating in nuclear power production.

This work is intended to be a contribution to the revaluation on a new basis, accommodated to the recent circumstances, of thermal nuclear power plants.

2. Earlier and recent considerations in the evaluation of thermal power plants

In addition to technical and economical consideration, the developments of thermal power plant types operating at present have been determined by different factors such as military considerations and fuel supply strategy, especially in the initial phase. Later as peaceful uses advanced, the importance of economy increased, and the main point in the development of the most promising types was to produce electricity at the possible lowest cost. Thus three competitive types of thermal power plants have widely spread by the seventies such as power plants with light-water reactors, heavy-water reactors, and gas-cooled reactors.

The basis for comparison and evaluation of these types previously were engineering and technological aspects, safety, and investment, as well as production costs associated; the problems of fuel demand, fuel utilization, and fuel cycle were at that time overshadowed by the low uranium costs. Also, there were much less nuclear power plants in operation at that time, and the estimates concerning the contribution of fast breeder reactors predicted a much sooner industrial-scale realization of the breeding process.

However, it has turned out recently that not before 10 to 15 years can be the breeding techniques mastered and thus a full-scale operation of breeder reactors be expected because of different difficulties in technology. This prediction, which seems somewhat pessimistic, is based on the opinion of many experts who no longer consider the liquid-metal cooled power plants — the development of which consumed most expenditure in both costs and effort — suitable. Namely, due to the technological difficulties encountered, the investment costs may increase to an extent never rendering such power plants economic according to some estimates. It is, therefore, right to suppose that, in the next ten years, other breeder reactors types will be preferred and much has to be done to make up for the time lost in the development of these types.

Since, due to the increasing energy demand, the utilization of nuclear power can not be given up in this intermediate period either, the number of nuclear power plants operating with thermal reactors will most likely increase beyond the number of such power plants being now under construction. It is, however, well known that the energy of uranium is exploited only to a very small extent (0.3 to 0.7%) by the present thermal power plants, and also the secondary fissile material production by the present thermal reactors is very low (varying in the range of 50 to 350 kg Pu [E]/[GW(e)]). The poor fuel efficiency can be, among others, fundamentally attributed to the fact that not only the operation of the thermal power plants already working but, in case of a nuclear power plant system of increasing capacity, also the initial fuel charge of the new power plants to be put into operation require large amounts of natural uranium, a fact that has not been taken into consideration in general in the earlier estimates which were based on the investigation of one power plant in isolation instead of a power plant system. Needless to say that, with this so-called engaged reserve neglected, the natural uranium demand of nuclear power engineering can be predicted rather erroneously. We called attention to this fact almost twenty years ago, and recommended at the same time a more realistic method of evaluation (1, 2). However, probably because of the minor importance of the utilization of nuclear power at that time, our warning did not meet the expected response.

However, situation has changed since. Recent estimates taking also the engaged reserve properly into consideration in compliance with our recommendation warn that, by the end of this century, the natural uranium demand of thermal power plants can be almost certainly met only by uranium exploited at much higher costs as compared with the uranium exploitation processes considered today economic. Therefore, one may reckon with a gradual increase of the uranium price in the future and also with temporary difficulties in fuel supply even if the fast reactors could be put into service within the said period with also the fuel reprocessing capacity developing accordingly.

All this warning has contributed to the present policy according to which, in addition to earlier considerations, also the specific natural uranium demand is taken into consideration in the evaluation of thermal power plants and in development strategy.

This work has been compiled on the basis of our earlier works (1 to 3) published in 1967 and 1971 to contribute to a revaluation of different types of thermal nuclear power plants on the basis of a more comprehensive system of considerations. In the next chapters, the specific natural uranium demand of the three most widely used thermal power plant types will be investigated in the light of different conditions. The investigations will conspicuously justify the recent trend of technical development where efforts have been made to accommodate certain thermal power plant types to the requirements of the once-through fuel cycle expectable in the long run.

3. Present thermal power plants

Before discussing the different types of nuclear power plants, it seems reasonably to briefly deal with the problem responsible for the poor utilization of uranium in all the three thermal power plant types today, which also sets a special direction to the technical development of this type of power plants.

3.1. *Once-through fuel cycle and fuel recycle*

In spite of the fact that the thermal power plant types operating at present have been designed almost without exception for fuel recycle, the majority of nuclear power plants is operated with once-through fuel cycle for the time being. This means that the spent fuel of the reactors is disposed after temporary in-plant storage in external deposits for indefinite time. Long-term storage is now a compulsion but it has certain advantages as well in that it offers the possibility of later choice between the reprocessing techniques not properly tested yet. As a matter of fact, these tactics result in increasing demand for storage capacity and wasting uranium utilization. To reduce the costs of the reprocessing methods which are rather expensive at present, the unit capacity of the reprocessing plants shall be increased to the appropriate extent first. According to calculations, the construction of a reprocessing plant will be profitable if it supplies a nuclear power system of a total capacity of 30 to 50 GW(e).

Since, however, the world's reprocessing capacity will fall short of the demand for at least 10 to 15 years, the number of countries where a reappraisal of fuel cycle of thermal power plants from the originally planned fuel recycle to once-through fuel cycle takes place is increasing. This means that all the technical and economical parameters which may affect the fuel cycle are so optimized that, assuming once-through cycle for the next ten to twenty years, both the unit electricity production costs and the natural uranium demand will be minimum.

The Tables for the light-water, heavy-water, and gas-cooled nuclear power plants discussed below contain such reoptimized technical parameters.

3.2. *Nuclear power plants operating with light-water reactors*

Two basic types of the recent generation of nuclear power plants operating with light-water reactors are known such as two-circuit system operating with pressurized-water reactor (PWR) and one-circuit system operating with boiling-water reactor (BWR). The share of such power plants in the total nuclear power production amounts to about 85%. Light-water power plants are technically well established proven systems and their construction permits meeting the strictest of environmental and safety requirements. Owing to these advantages, light-water nuclear power plants are now cheaper electricity producers than fossile-fired conventional power plants on a wide scale of the economic requirements. Light-water power plants constructed today are designed for an operating life of 25 to 30 years in general.

Today units of a capacity of 1300 MW(e) are produced of both PWR and BWR power plants but, according to literature, nuclear power plants of a unit capacity of 400 to 600 MW(e) will be similarly important for a long time in the future because of the low gross capacity of the electrical network of small countries and developing countries. Table 1 gives the most important technical characteristics and fuel consumption figures for the LWR power plant types to be, or already having been, standardized in different countries. Western countries have decided on producing units of a capacity of about 1300 MW(e) while in the USSR, VVER-1000 and RBMK-1000 types have been standardized. The VVER-440 pressurized-water reactor has been chosen to represent lower capacity units in our comparison, the majority of the tabulated data being adopted from the Reports (5, 6) of the large-scale international research programme on International Nuclear Fuel Cycle Evaluation (INCFE) launched by the International Atomic Energy Agency in 1978, and finished two years later (1980).

As seen from the tabulated data, there is little difference between the West-European types and those developed in the USA in respect of the technical parameters of fuel consumption. For instance, for equilibrium refuelling, uranium enriched to 3.1% is used for PWRs while that with an enrichment of 2.8% for BWRs, the equilibrium enrichment of VVER being 4.5%. In the western countries, the burnup of PWRs is 30,000 to 33,000 MWday/t on the average while the burnup of BWRs ranges from 27,000 to 30,000 MWday/t. In the Soviet VVER-1000 nuclear power plants, a burnup of 40,000 MWday/t is planned. In respect of the specific natural uranium demand of the initial core and equilibrium refuelling, the French and American PWRs are most economic, the uranium demand of VVER-440 being somewhat higher according to data available from the USSR. This higher uranium demand of VVER-440, resulting from the somewhat higher uranium enrichment factor of this type, may be compensated to some extent by the relatively higher ratio of fissile uranium remaining in the spent fuel, and/or of the plutonium produced. However, this advantage will be appreciable only after spent fuel is reprocessed on an industrial scale.

Recent increasing efforts to improve the fuel utilization of light-water power plants indicate that the warning inherent in the predictions concerning future uranium supply has not been ineffective and, on the other hand, once-through fuel cycle will predominate under compulsion in the long run. The most important practices explicitly outlined in the literature are, as follows:

— increased burnup
— lattice changes
— spectrum shift
— enrichment zoning (including blankets)
— full use of early batches of startup core
— reconstitution/inversion of BWR fuel
— end-of-cycle power coastdown.

Development experts do hope that no LWR systems considerably differing from those used at present will be the result of such modifications. Savings in uranium expected to be achieved by means of these methods range from 2 to 12%. Detailed description of the different methods, accurate savings estimates, and the impacts on costs can be found in the large number of reports (5, 7 thru 9) on this subject. Note that there is some interrelation between the different improvement methods so that the contributions to a cumulative reduction in uranium requirements would not be additive. Since increased burnup is one of the most promising methods, the parameters calculated for increased burnup for both a PWR and a BWR power plant are also given in Table 1.

Table 1
Fuel consumption figures of nuclear power plants operating with LWR

Parameters \ Types	PWR						BWR		PWR with increased burnup	BWR with increased burnup
	France	Germany F. R.	USA	USSR VVER-1000	USSR VVER-440	Norway	USA	USSR RBMK-1000	USA	USA
Total thermal power [MW]	3817	3765	3800	3000	1375	2790	3800	3200	3800	3800
Electrical power										
Gross	—	1299	1344	1000	440	—	1344	1000	1344	1344
Net	1300	1229	1270	945	408	950	1270	—	1270	1270
Net efficiency [%]	34	32.6	33.4	31.5	29.7	34	33.4	31.3	33.4	33.4
Fraction of core replaced (refuelling)	0.33	0.33	0.33	—	0.33	0.23	0.22	—	0.20	0.13
Refuelling interval [years]	1.07	1	1.07	—	1	1	1	—	1.07	1
Equilibrium reload enrichment [%]	3.1	3.15	3.0	4.52	3.6	2.721	2.91	1.8	4.3	3.79
Average discharge burnup [MW day/t]	31 800	33 000	30 390	40 000	28 600	27 500	25 400	18 100	50 650	47 000
Peak pellet burnup [MW day/t]	47 000	48 000	—	—	—	—	40 000	—	65 000	76 000

Natural uranium requirements[1] [t/GW(e)]										
Initial core	324	367	303	—	435	430	349	—	314	
Annual equilibrium reload	140	139	139	171	211	132	146	—	123	116
30-year cumulative										
Gross	4380	4363	4347	—	—	4314	4610	—	3823	3825
Net[2]	—	4224	4193	—	—	—	—	—	3691	—
Fissile material in spent fuel[1]										
Enrichment [Wt % H. E.]										
^{235}U	0.85	0.82	0.85	1.26	1.31	0.77	0.93	—	0.78	0.66
Pu-fiss	0.68	0.66	0.68	0.74	0.69	0.57	0.64	—	0.81	0.77
Annual equilibrium discharge [kg/GW (e)]										
^{235}U	202	188	205	275	365	206	257	—	109	110
Pu-fiss	168	152	166	161	192	153	169	—	112	121

Note:
[1] Normalized to 70% capacity factor and 0.2% tails enrichment
[2] 30-year cumulative requirements are 30 year gross cumulative requrements less credit for partially burned last-core fuel

3.3. *Nuclear power plants operating with heavy-water reactor*

For the time being, there are two main development lines for nuclear power plants operating with heavy-water reactor — one being the CANDU concept using pressure tubes to cool the core within a nearly unpressurized moderator tank while the other is the pressure vessel design from the Federal Republic of Germany, in which coolant channels and the surrounding moderator are held at equal pressure. One of the incentives of heavy-water power plants is that they can be operated also with natural uranium so that their fuel cycle is independent of the enrichment capacity, a fact which makes this type of power plant especially attractive in many countries.

A number of CANDU reactors have been in operation successfully in several countries. Highest unit capacity is 750 MW(e) at present but design studies of 1000 to 1200 MW(e) HWR nuclear power plants have been completed as well. The first industrial-scale pressurized vessel-type HWR power plant was finished in Federal Republic of Germany in 1974 and it operated reliably since. This design can be adapted to a gross electrical output of 685 MW(e) without conceptual changes or major problems. To the design of still larger units, limits are set by the unsettled state of the production technology of largesize reactors vessels.

The most important technical parameters and fuel consumption figures for the advanced versions of the two HWR power plant types are tabulated in Table 2. As seen, there is little difference between the fuel utilization figures.

In addition to the difference already mentioned, heavywater power plants differ from the light-water systems in the following important aspects:

— In HWR power plants, the primary system components are of larger size. HWR power plants require large amounts of heavy water for their operation, and an auxiliary system for heavy water handling and upgrading. Due to these factors, the investment costs of HWR power plants lie above the investment cost of LWR power plants.

— On-power refuelling is used with the heavy-water reactors, resulting in a by about 5% higher capacity factors as compared with the light water systems where batch refuelling is used.

— The specific plutonium production of HWR power plants fuelled with natural uranium is higher than that of the LWR power plants, however, the concentration of plutonium in the spent fuel of HWRs is lower. Fuel through put i.e. the amount of fuel removed from, and introduced into, the reactor for HWRs is about four times as much as for LWRs, the activity of spent fuel being, however, lower.

From among the methods to improve the uranium utilization of HWR power plants, that substituting low-enriched uranium for natural uranium fuel seems most promising. Investigations showed that, below an enrichment of

Table 2

Fuel consumption figures of nuclear power plants operating with HWR

Parameters / Types	Natural uranium fuelled nuclear power plants		Low-enriched uranium fuelled nuclear power plants		
	Canada	Germany F. R.	Canada	Germany F. R.	USA
Total thermal power [MW]	3425	2160	3425	2160	4029
Electrical power [MW]					
Gross	1074	685	1074	685	1343
Net	1000	637	1000	637	1260
Net efficiency [%]	29.2	29.2	29.2	29.5	31.5
Fuel residence time [epfd]	276	242	790	654	813
Equilibrium reload enrichment [%]	0.711	0.711	1.2	1.2	1.2
Average discharge burnup [MW day/t]	7300	7400	20 900	20 000	19 750
Peak pellet burnup [MW day/t]	—	—	—	—	25 000
Natural uranium requirements[1] [t/GW(e)]					
Initial core	130.8	112	256	150	133.1
Annual equilibrium reload	121.1	118.4	82.3	86.1	82
30-year cumulative					
Gross	3716	3608	2651	2679	2559
Net[2]	—	3552	—	2569	2466
Fissile material in spent fuel[1]					
Enrichment [Wt % H. E.]					
^{235}U	0.23	0.21	0.1	0.074	0.1
Pu-fiss	0.276	0.253	0.344	0.314	0.347
Annual equilibrium discharge [kg/GW(e)]					
^{235}U	276	245	41.7	31.5	41.4
Pu-fiss	331	296	140	133	139

Note:

[1] Normalized to 70% capacity factor and 0.2% tails enrichment

[2] 30 year cumulative requirements are 30 year gross cumulative requirements less credit for partially burned last-core fuel

1.2%, both the reactor design and the fuel assembly construction needed only insignificant modification. This enrichment ratio is high enough to include the optimum not only from a technical but, also from an economical point of view. Table 2 gives also the technical characteristics and fuel consumption figures of HWRs fuelled with low-enriched uranium. On the basis of the data on natural uranium requirement, it can be seen that not only equilibrium natural uranium requirement but also the 30 years gross cumulated natural uranium requirement is kept very low by the enrichment.

3.4. *Nuclear power plants operating with high-temperature gas-cooled thermal reactors*

HTGR power plants can be considered today a relatively unsettled type in spite of the experiments going back far to the past.

The reactors of such power plants are graphite-moderated, helium-cooled systems, achieving a very favourable neutron economy due to the negligible neutron absorption by the mentioned materials and to the unnecessity of metal structural elements in their core. Characteristics of HTGR are also the low power density, the high heat capacity of the core, and the safety due to the negative temperature of the coolant permits achieving a high thermodynamic efficiency in both electricity production and direct industrial applications. The fuel of gas-cooled reactors consists in general of small coated particles, which contain fissile and/or fertile material.

Only one advanced prototype of helium-cooled nuclear power plant is in operation at present: the nuclear power plant in Fort St. Vrain, USA, with a capacity of MW(e). Another such type of power plant the THTR-300 in Federal Republic of Germany is now under construction. In the gas-cooled nuclear power plant technology, the eighties and nineties are intended to be a demonstration period and construction of the first large-scale gas-cooled nuclear power plants is not expected before the end of this century provided the development will have been successful.

In gas-cooled thermal reactors, also the thorium (^{232}Th—^{233}U) fuel cycle is considered feasible and much research is done in this field in many places. Table 3 gives only the technical parameters and fuel consumption figures of gas-cooled systems based explicitly on uranium-plutonium fuel cycles. As seen from the Table, very high burnups are planned in HTGR power plants with a view to favourable uranium utilization and for the sake of economy.

Since the HTGR concepts differ from each other considerably not only in reactor design but also in fuel geometry and fuel handling, also the variance in their natural uranium requirement cumulated over 30 years is greater than for LWR and HWR power plants. The average uranium requirement is rather

Table 3

Fuel consumption figures of nuclear power plants operating with HTGR

Parameters / Types	Japan	USA	Germany F. R.	Great Britain	France
Total thermal power [MW]	3000	3360	3000	2336	3000
Electrical power [MW]					
Gross	—	1360	—	952	—
Net	—	1332	1240	916	1170
Net efficiency [%]	40	39.6	41.0	39.2	—
Fuel residence time [years]	3	3	3.07	3.26	4
Equilibrium reload enrichment [%]	6	10.1	8.5	7.9	12
Average discharge burnup [MW day/t]	63 000	111 000	100 000	100 000	127 000
Natural uranium requirements[1] [t/GW (e)]					
Initial core	345	186	178	225	—
Annual equilibrium reload	114	118	100	98	—
30-year cumulative					
Gross	3550	3533	3147	3177	3850
Net[2]	—	3446	3096	—	—
Fissile material in spent fuel[1]					
Enrichment [Wt %]					
$^{235}U/U$	1.4	1.3	1.6	1.05	2.4
Pu-fiss/Pu	60	56	56	49	64.2
Annual equilibrium discharge [kg/GW(e)]					
^{235}U	131	58	84	60	128
Pu-fiss	79	45	57	46	86

Note:
[1] Normalized to 70% capacity factor and 0.2. % tails enrichment
[2] 30-year cumulative requirements are 30 year gross cumulative requirements less credit for partially burned last-core fuel

favourable: it lies in the vicinity of the requirement of heavy-water power plants fuelled with natural uranium. Because of the relatively high uranium enrichment factor, the specific amount of plutonium recoverable from the spent fuel falls, however, short of that of the other thermal types.

4. Determination of the specific natural uranium demand of nuclear power systems operating with thermal reactors

All what will be said below, including the derivation, has already been said in our earlier works (1 thru 3), with only little modification being introduced now. These modifications were introduced first in our work referred to under (4).

The equilibrium natural uranium demand of a thermal nuclear power plant can be determined on the basis of the amount of fresh fuel introduced on refuelling and of the level of enrichment in the following way:

$$G = \frac{m_t \cdot p_{cs}}{T_{cs}} \cdot \frac{d - d_{sz}}{d_t - d_{sz}} \qquad \left[\frac{\text{mass}}{\text{time}}\right] \tag{1}$$

where

m_t [mass] mass of fuel in reactor
p_{cs} [1] part of core replaced on refuelling
T_{cs} [time] average time interval between refuellings
d [1] average enrichment of fresh fuel introduced (^{235}U content)
d_{sz} [1] ^{235}U content of depleted uranium (enrichment tails)
d_t [1] ^{235}U content of natural uranium.

The specific energy yield of thermal power plants i.e. the net electricity as compared with the amount of natural uranium used is defined by the following relationship:

$$H = \frac{P \cdot L}{G} \qquad \left[\frac{\text{energy}}{\text{mass}}\right] \tag{2}$$

where

P (power) net electric output
L [1] capacity factor.

If a thermal nuclear power plant is operated with fuel recycle i.e. the recovered fissile material content of the spent fuel (^{235}U, ^{239}Pu, ^{241}Pu) is used for the enrichment of the fuel to be used on refuelling, then, neglecting any fuel loss and assuming the fissile materials of the same amount but of different kinds to be identical as far as their contribution to reactivity is concerned, the specific energy yield of the nuclear power plant can be calculated, as follows:

$$H_{\text{closed}} = \frac{H}{1 - \dfrac{U_{\text{out}} + \text{Pu}_{\text{out}}}{\text{U}_{\text{in}}}} \qquad \left[\frac{\text{energy}}{\text{mass}}\right] \qquad (3)$$

where

U_{out} [mass/time] mass of ^{235}U removed with spent fuel from the reactor, related to unit time

Pu_{out} [mass/time] combined mass of ^{239}Pu and ^{241}Pu removed with the spent fuel from the reactor, related to unit time

U_{in} [mass/time] mass of ^{235}U introduced with the fresh fuel to the reactor, related to unit time.

For the parameters G, H, and $H_{\text{rec.}}$ defined above, only the natural uranium demand required for the operation of the nuclear power plant(s) has been taken into consideration. However, for a nuclear power system (consisting of thermal power plants) the gross capacity of which is increasing continuously, not only the fuel required to operate the existing nuclear power plants but also the initial fuel charge of the reactors of new power plants to be put into operation shall be provided, which occurs as an additional demand for natural uranium. Therefore, the specific energy yield of a nuclear power plant system of increasing capacity is less than that of one single nuclear power plant already operating.

Assuming the capacity of the nuclear power plant system to increase exponentially — this assumption can be considered still realistic today — the specific energy yield of the system can be determined as shown below.

Let the doubling time of capacity be T_2 [time]. With T_2, the time function of the capacity of the nuclear power plant system will be:

$$P(t) = P_0 \cdot e^{ct} = P_0 \cdot e^{\frac{\ln 2}{T_2} t} \qquad [\text{power}] \qquad (4)$$

where

P_0 [power] capacity associated with $t = 0$,

c [1/time] growth parameter, $c = \ln 2 / T_2$.

Assuming that the capacity factor (L) will not change in time, the system produces net electricity

$$E(t_1, t_2) = L \int_{t_1}^{t_2} P(t)dt = \frac{L \cdot P_0}{c} (e^{ct_2} - e^{ct_1}) \qquad [\text{energy}] \qquad (5)$$

in interval t_1 to t_2. This required natural uranium of a mass of

$$\frac{E(t_1, t_2)}{H}.$$

Within the same period, the growth of the capacity of the system will be

$$\Delta P(t_1, t_2) = \int_{t_1}^{t_2} \frac{dP(t)}{dt}\, dt = \int_{t_1}^{t_2} cP_0 e^{ct} dt =$$

$$= P_0(e^{ct_2} - e^{ct_1}) = \frac{c}{L} E(t_1, t_2) \qquad \text{[power]} \qquad (6)$$

In the knowledge of the amount of natural uranium required for the initial charge of the power plant type of which the system is composed and/or of the specific value of this amount of uranium as related to capacity (m_r[mass/power]), then the amount of natural uranium required for the initial charge of the new power plants put into service in interval t_1 to t_2 can be calculated using the following relationship:

$$m_u(t_1, t_2) = \Delta P(t_1, t_2) \cdot m_r \qquad \text{[mass]} \qquad (7)$$

Considering that processing of the natural uranium coming from the uranium mine or other deposit into fuel requires a certain time τ [time], natural uranium required for the growth in capacity of a period to which τ is added $(t_1 + \tau, t_2 + \tau)$ shall be provided in interval (t_1, t_2) i.e. natural uranium of a mass of

$$m_u(t_1 + \tau, t_2 + \tau) = \Delta P(t_1 + \tau, t_2 + \tau) \cdot m_r =$$

$$= P_0 e^{c\tau}(e^{ct_2} - e^{ct_1}) \cdot m_r = = \frac{c \cdot e^{c\tau}}{L} E(t_1, t_2) \cdot m_r \quad \text{[mass]} \qquad (8)$$

Hence, the specific energy yield (F[energy/mass]) of the nuclear power plant in given period:

$$F(t_1, t_2) = \frac{E(t_1, t_2)}{\dfrac{E(t_1, t_2)}{H} + \dfrac{ce^{c\tau} \cdot m_r \cdot E(t_1, t_2)}{L}} =$$

$$= \frac{H}{1 + \alpha H} \qquad \left[\frac{\text{energy}}{\text{mass}}\right] \qquad (9)$$

where

$$\alpha = \frac{ce^{c\tau}}{L} \cdot m_r = \frac{\frac{\ln 2}{T_2} \cdot e^{\frac{\ln 2}{T_2} \cdot \tau_0}}{L} \qquad \left[\frac{\text{mass}}{\text{energy}}\right] \tag{10}$$

that means that F is independent of the chosen interval. It is worth mentioning that this applies to systems of exponential growth only. In case of a fuel recycle, the relationship given below will apply:

$$F_{\text{rec.}}(t_1, t_2) = \frac{H_{\text{rec.}}}{1 + \alpha H_{\text{rec.}}} \qquad \left[\frac{\text{energy}}{\text{mass}}\right] \tag{11}$$

5. Evaluation of the types of thermal nuclear power plants on the basis of their specific natural uranium demand within the nuclear power system

Table 4 gives the parameters of the different types of thermal nuclear power plants, which are required to determine the specific natural uranium demand. In the Table, specific values $\bar{G}$, $\bar{U}_{out}$, $\bar{Pu}_{out}$ and $\bar{U}_{in}$ related to unit capacity have been indicated in place of quantities G, U_{out}, Pu_{out} and U_{in} defined previously, and a capacity factor of $L=0.7$ as well as an enrichment tails of $d_{sz}=0.002$ had been used uniquely in determining these values.

Table 5 shows the result of calculations concerning specific energy yield of different thermal nuclear power plants in a power system. Specified in the Table are the reciprocal of the specific energy yield of power plant systems i.e. the specific natural uranium demand (g [mass/energy]), as well as its value related to the total energy content of natural uranium i.e. the so called material efficiency (η) for both once-through cycle and recycle. In the calculation, the doubling time of capacity (T_2) was assumed as 10 years while the time required to produce the fuel (τ) as 0.5 years. Total energy content of natural uranium was assumed to be 830 000 MWday/t.

Relationship (9) expresses the change of the specific energy yield of the nuclear power plant system as a function of the rate of growth of the capacity. It can be seen that the higher the rate of growth, the smaller is the specific energy yield because the major part of natural uranium is used for the initial charge in systems where the capacity increases rapidly. Since in the range of reasonable growth rates the specific energy yield may change considerably, we found interesting to illustrate these relationships also graphically. Figs. 1 thru 4 show the change of specific energy yield of different types of nuclear power plants in a power system as a function of doubling time of capacity. In the Figures, also a scale to read the material efficiency has been given on the right.

Table 4

Data used to determine the specific energy yield (material efficiency) of thermal nuclear power plants operating within the nuclear power system (for definition of the tabulated quantities see Chapter 4)

Type of nuclear power plant		$\bar{G}$[1] [t/year/GW (e)]	$\bar{U}_{out}$[1]	$\overline{Pu}_{out}$[1] [kg/year/GW (e)]	$\bar{U}_{in}$[1]	m_r [t/GW (e)]
PWR	France	140	202	168	764.7	324
	Germany F. R.	139	188	152	7584	367
	USA	139	205	166	761.0	303
	USSR VVER-1000	171	275	161	914.3	—
	USSR VVER-440	211	365	192	1142	435
BWR	Norway	132	206	153	728.0	430
	USA	146	257	169	801.1	349
PWR with increased burnup	USA	123	109	112	659,2	314
BWR with increased burnup	USA	116	110	121	625.8	382
HWR	Canada CANDU	121.1	276	331	861.0	130.8
	Germany F. R. PVHWR	118.4	245	296	841.8	112
HWR lowenriched uranium fuelled	Canada CANDU	82.3	40.7	140	504.7	256
	Germany F. R. PVHWR	86.1	31.5	133	528.0	150
	USA	82	41.4	139	502.8	133.1
HTGR	Japan	114	131	79	602.6	345
	USA	118	58	45	615.3	186
	Germany F. R.	100	84	57	523.3	78
	Great Britain	98	60	46	513.8	225

Note:
[1] Normalized to 70% capacity factor and 0.2% tails enrichment

Table 5

Results of calculations for the specific energy yield (material efficiency) and natural uranium demand of thermal nuclear power plants operating within the nuclear power system (for definition of the tabulated quantities see Chapter 4)

Type of nuclear power plant		Once-through cycle				Recycle			
		H [MWday/t]	F [MWday/t]	η [%]	g [kg/MWday]	H [MWday/t]	F [MWday/t]	η [%]	g [kg/MWday]
PWR	France	1825	1635	0.197	0.612	3536	2886	0.348	0.346
	Germany F. R.	1838	1623	0.195	0.616	3332	2686	0.324	0.372
	USA	1838	1656	0.200	0.604	3587	2955	0.356	0.338
	USSR VVER-1000	1494	—	—	—	2856	—	—	—
	USSR VVER-440	1211	1097	0.132	0.911	2365	1967	0.237	0.508
BWR	Norway	1936	1663	0.200	0.601	3819	2887	0.348	0.346
	USA	1750	1562	0.188	0.640	3737	2974	0.358	0.336
PWR with increased burnup	USA	2077	1841	0.222	0.543	3125	2619	0.316	0.382
BWR increased burnup	USA	2203	1890	0.228	0.529	3491	2766	0.333	0.361
HWR	Canada CANDU	2110	2001	0.241	0.500	7151	6040	0.728	0.166
	Germany F. R. PVHWR	2158	2060	0.248	0.485	6039	5330	0.642	0.188
HWR low-enriched uranium fuelled	Canada CANDU	3104	2685	0.323	0.372	4836	3889	0.469	0.275
	Germany F. R. PVHWR	2967	2729	0.329	0.366	4311	3824	0.461	0.261
	USA	3116	2881	0.347	0.347	4859	4310	0.519	0.232
HTGR	Japan	2241	1945	0.234	0.514	3440	2789	0.336	0.358
	USA	2165	2006	0.242	0.499	2601	2374	0.286	0.421
	Germany F. R.	2555	2345	0.283	0.426	3497	3116	0.375	0.321

Note: Taking $T_2 = 10$ years, $\tau = 0.5$ year and, for the total energy content of natural uranium 830000 MW day/t into consideration.

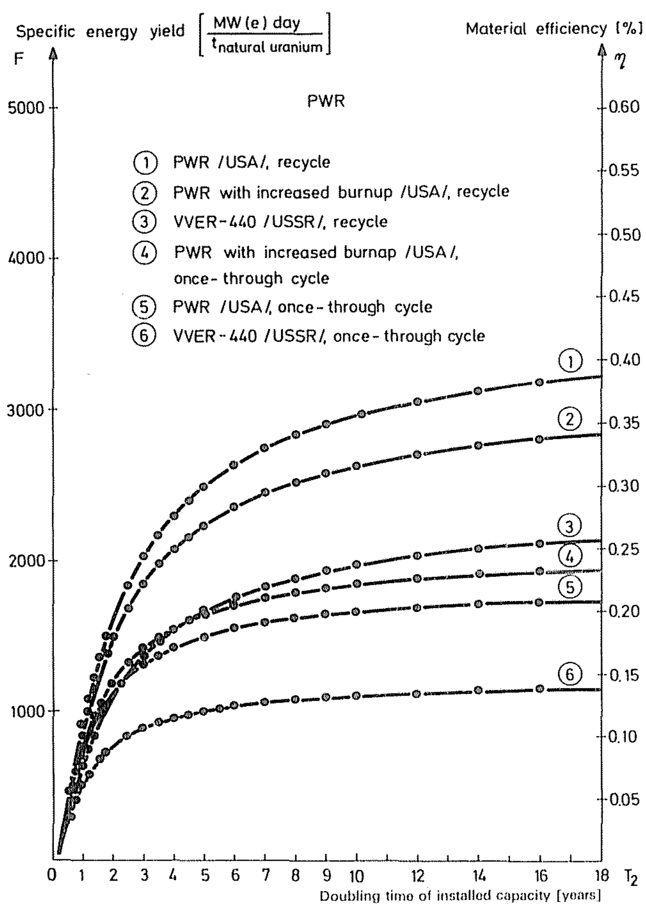

Fig. 1. Specific energy yield and material efficiency of nuclear power systems operating with PWRs as a function of doubling time of installed capacity

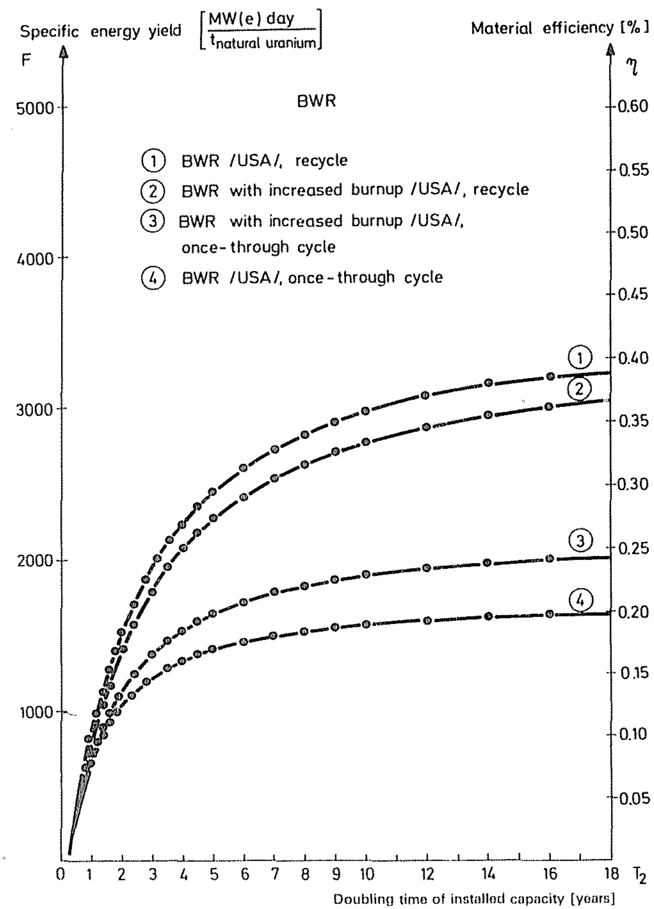

Fig. 2. Specific energy yield and material efficiency of nuclear power systems operating with BWRs as a function of doubling time of installed capacity

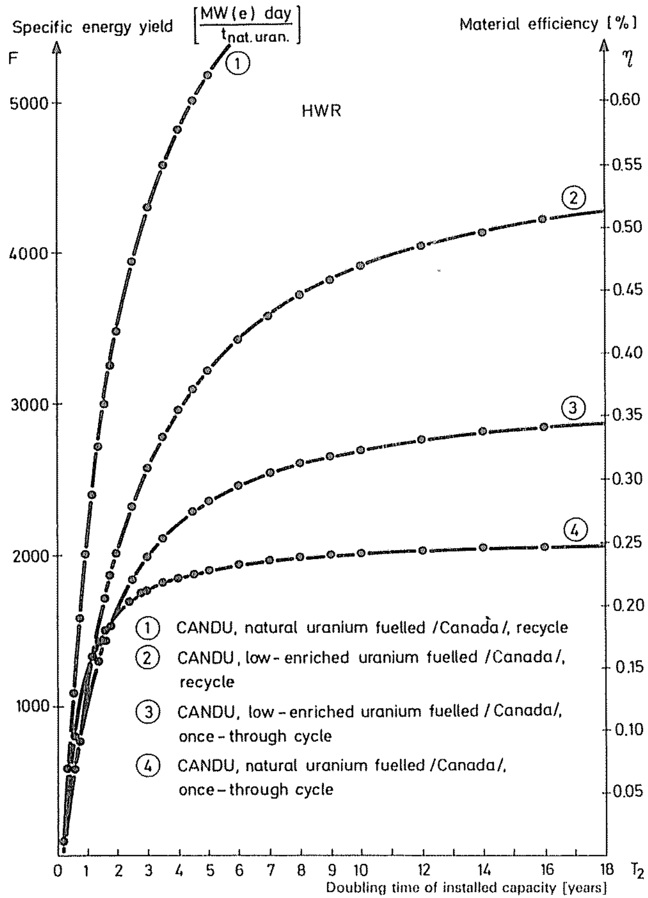

Fig. 3. Specific energy yield and material efficiency of nuclear power systems operating with heavy-water (CANDU) reactor as a function of doubling time of installed capacity

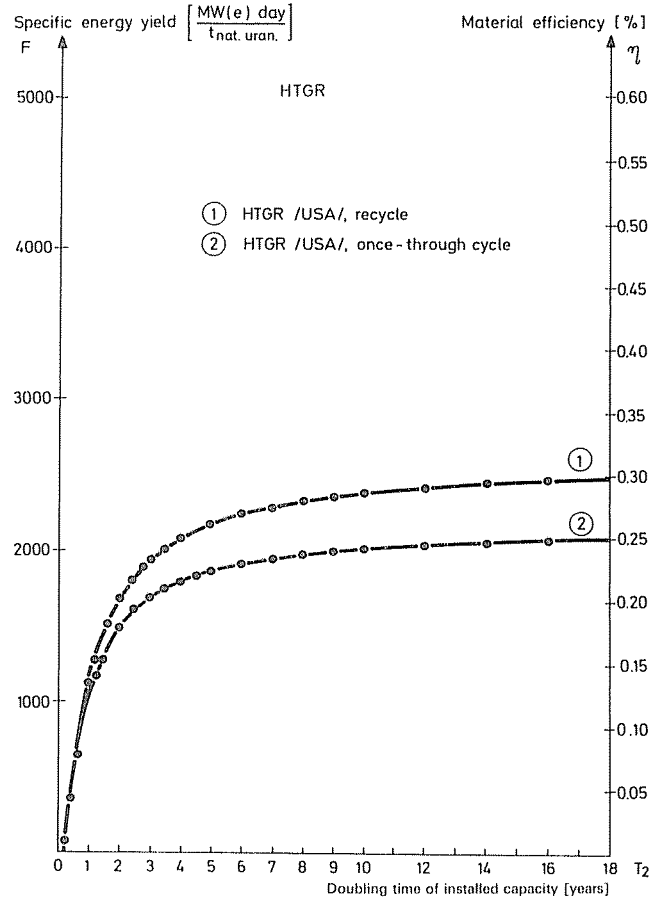

Fig. 4. Specific energy yield and material efficiency of nuclear power systems operating with HTGRs as a function of doubling time of installed capacity

An analysis of the curves leads to the following remarkable conclusions concerning uranium utilization of thermal nuclear power plants:

— In case of the reference PWR, increase of the burnup (which is planned in order to improve the material efficiency) actually improves the efficiency for once-through cycle but reduces it for recycle (although here much higher efficiencies are involved). This points first of all to the fact that the increase of burnup was optimized on the assumption of an once-through cycle. The same applies to the reference BWR (Figs 1 and 2).

— In case of nuclear power plants with CANDU reactor, similar results are obtained when low enriched uranium is used with recycle. In case of once-through cycle, the power plants fuelled with low-enrichment uranium have higher efficiency while with recycle, the material efficiency of natural uranium fuelled power plants is higher (although the efficiency increases considerably). This again indicates that enrichment has been optimized for once-through fuel cycle.

— There is little difference between PWR and BWR reference power plants while, in respect of material efficiency, the VVER-440 is fairly inferior to them. This proves that units of lower capacity are less economic also in respect of fuel utilization and, on the other hand, indicates that VVER-440 has not been modernized yet in the way in which the BWR and PWR referred to had been, or were planned to be, modernized.

— The material efficiency of nuclear power plants with CANDU reactor is much superior to all the other types for both once-through cycle and recycle.

— HTGR power plants are most insensitive to whether the cycle is once-through cycle or recycle as far as material efficiency is concerned. In case of once-through cycle, their material efficiency is higher than that of light-water plants.

The fact that the above conclusions concerning optimization for once-through cycle are confirmed also in the references (7 thru 10) proves the feasibility of our relationships published earlier (1, 2, 3) and also here under Chapter 4 and it also proves that all what has been said here is a realistic approach to the problem.

Summary

The long standing clarification of breeding techniques in connection with the shortage of existing reprocessing capacities require rigorous conservation of natural uranium resources. The present paper is intended to contribute to the re-evaluation of competitive types of thermal reactors and power systems based on these reactors according to the changing recent and future fuel availability. The mathematical model established takes into account the natural uranium demand of operating reactors as well as the initial fuel charge of new power plants. The specific natural uranium demands are considered as the function of the growth of installed nuclear capacities. The investigations are extended to the once-through fuel cycle modes and the fuel recycle modes too.

References

1. Чом, Д.: Некоторые вопросы топливого цикла ядерных реакторов. Теплоэнергетика, 1967/9, 85—88.
2. Чом, Д.: Иследование сустем атомных электростанций с тоски зрения комплексного исползования запасов ядерного топлива. Симпозиум СЭВ, Обнинск, 1967.
3. Lévai, A.—Csom, Gy.: Realistic Evaluation of Nuclear Fuel in Atomic Energy Systems. — The Eight World Energy Conference, Bucharest, 1971—2.4—51.
4. Csom, Gy.—Fehér, S.: Characteristics of the Different Types of Nuclear Power Plants, Relevant in Respect of Fuel Cycle. — BMR-TR (Training Reactor, Technical University Budapest) 113/81, Budapest, 1981.
5. Advanced Fuel Cycle and Reactor Concepts. — Report of INFCE WG 8. IAEA, Vienna, 1980.
6. INFCE Summary Volume, IAEA, Vienna 1980.
7. Improvement of Nuclear Fuel Management in PWRs. France, July 1978, INFCE/DEP/WG. 8/10.
8. Potential Improvements to the Once-Through Fuel Cycle for LWRs. Germany, F. R., 30 June 1979, INFCE/DEP/WG. 8/18.
9. Improved Once-Through Fuel Cycles for LWRs. USA, 30 November 1978, INFCE/DEP/WG. 8/80.
10. Data Base for a CANDU-PHW Operating on a Once-Through, Slightly Enriched Uranium Cycle. Canada, October 1978, INFCE/DEP/WG. 8/7.

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