

Nuclear power - the energy balance

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Note

In this document the references are coded by Q-numbers (e.g. Q6). Each reference has a unique number in this coding system, which is consistently used throughout all publications by the author. In the list at the back of the document the references are sorted by Q-number. The resulting sequence is not necessarily the same order in which the references appear in the text.

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Part B

The reference reactor

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B1 Which technology?

The only fissile nuclide in nature is uranium-235, accounting for 0.71% of the atoms in the uranium as found in nature (called: natural uranium). The remaining 99.3% are uranium-238 atoms and traces of uranium-234, both of which are not fissile.

In an operating nuclear reactor a part of the abundant and 'fertile' uranium-238 in natural uranium is converted by neutron capture into plutonium-239, which is fissile.

There are two main classes of nuclear reactors: burners and breeders.

In a *burner reactor* no more than 0.6 - 0.7% of the atoms in the natural uranium leaving the mine can be fissioned. The conversion rate of U-238 into fissile plutonium nuclides is less than the rate at which fissile nuclides (U-235 and newly formed Pu-239) are fissioned. When the fissile content of the fuel in the reactor falls below a given value (about 0.8%), the fuel has to be replaced by fresh fuel.

In a *breeder reactor* more fissile nuclides - plutonium-239 and Pu-241 - are formed than are fissioned. Theoretically, some 30-60% of the atoms in natural uranium could be fissioned in this way in a breeder reactor. That would mean 50 to 100 times the fissioned amount in conventional burner reactors. Breeder reactors operate with fast neutrons and therefore are often called 'fast breeders', which does not mean the breeding process goes fast. Besides, not every fast reactor is a breeder (see also Figure B.1).

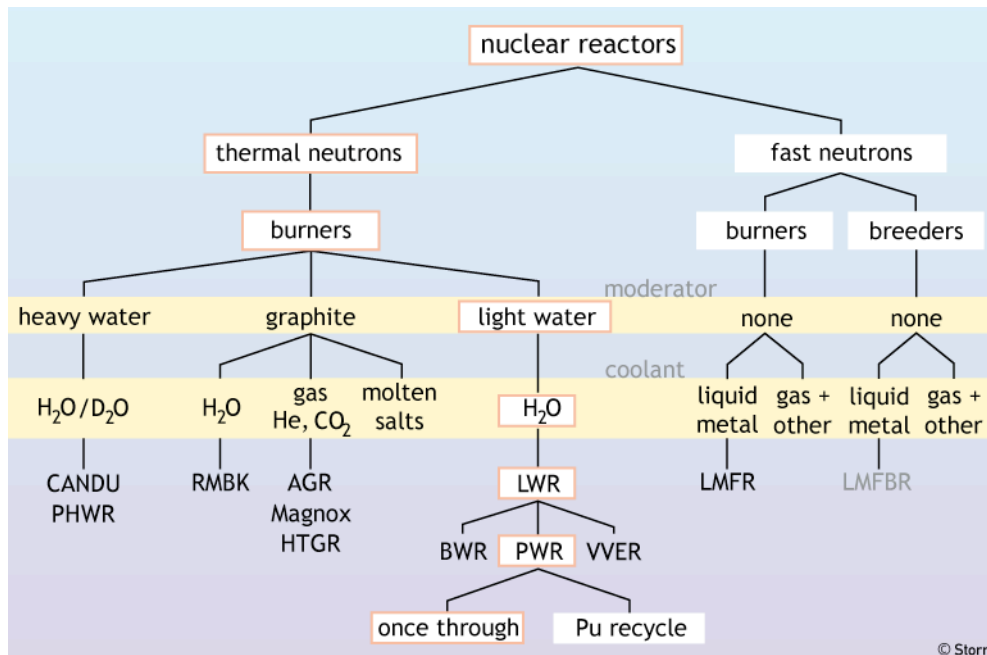


Figure B.1 Decision tree to define the reference nuclear reactor. Only that types of power reactors are mentioned at the bottom of the diagram that are actually operating or have been operating. The LMFB is not operational (see text). The LWR has been chosen as the reference reactor type in this study. The differences between the Boiling Water Reactor BWR, Pressurized Water Reactor PWR and the Russian VVER (also a PWR) with respect to the energy analysis are minor. The LWR with plutonium recycling is not included in this study.

Burner reactors

All power reactors currently operating are burner reactors, based on fission with thermal (slow) neutrons. The three main classes are (see also Figure B.1):

- light-water reactors LWR: Pressurized Water Reactor PWR and Boiling Water Reactor BWR,
- graphite-moderated reactors and gas-cooled reactors, e.g. Magnox and AGR
- heavy-water moderated reactors, e.g. CANDU.

At present 88% of the power reactors of the world are LWRs (see Figure B.2).

The reference reactor, which may serve as model for the newest currently operating power reactors, achieves a lifetime uranium utilization of less than 0.6% (see Table B.3). This implies that less than 6 grams of each kilogram natural uranium as delivered by the mine, actually are fissioned. The remaining 994 grams leave the nuclear energy system as depleted uranium and highly radioactive spent fuel.

Advanced reactors, such as the so-called Generation III reactors and the Pebble Bed Reactor PBMR, all are burner reactors and may reach a uranium utilization of slightly higher than 0.6%. One of the first Generation-III reactors, an EPR, is being built at Olkiluoto in Finland. As none of the advanced reactors are operating yet, no operational data are available and no uranium utilization figures can be calculated and therefore the figure of ‘slightly higher than 0.6%’ is speculative.

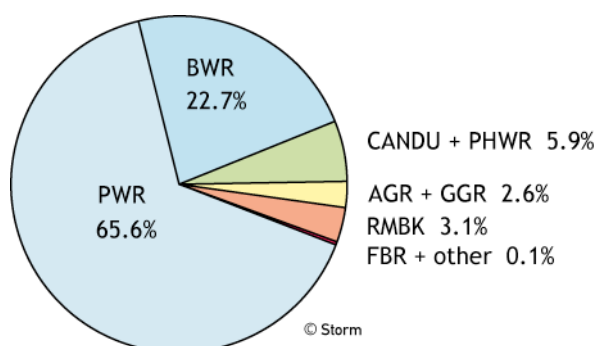


Figure B.2

Distribution of the world nuclear power reactors in 2007, according to power output. The section PWR includes the Russian version (PWR-VVER). 88.4% of the world nuclear capacity is supplied by light-water reactors (LWR) and 11.6% by other types.

Source: European Nuclear Society, www.euronuclear.org/info/.

Breeders

The high fissionable fraction of natural uranium theoretically achievable by the breeder is the source of the old nuclear dreams from the 1950s: ‘all nuclear society’, ‘too cheap to meter’ and ‘burning the rocks’. Today these unproven figures still give rise to the technical dreams of untold quantities of cheap, clean nuclear energy for all mankind for the next centuries.

However, there are two serious obstacles on the road to the materialization of these technical dreams, each of which is prohibitive:

- the technical unfeasibility of the breeder system
- the uncontrollable and high risk of plutonium terrorism and the proliferation of nuclear technology.

What is called a ‘breeder’ is not just a reactor type or a stand-alone system. To exploit fully the promised potential of natural uranium, a complex breeder cycle is a prerequisite, consisting of three components: a breeder reactor, a reprocessing facility and a fuel fabrication plant (see Figure B.3)

Important parameters of the breeder system are, among other:

- initial inventory of plutonium in Mg/GW, should be as low as possible
- breeding ratio, should be as high as possible
- full-power operating time of the reactor, should be as long as possible
- out-of-pile time of the plutonium, should be as short as possible
- plutonium losses in the cycle, should be as low as possible.

The first three are reactor parameters, the latter two are determined by two of the other components of the cycle: reprocessing and fuel fabrication.

All three components of the breeder cycle must operate flawlessly, continuously and exactly tuned to the other two components, in order to let the system actually breed more fissile material from non-fissile uranium-238 than it consumes. If one component fails, the whole system fails. In fact, none of the three components have ever demonstrated operation as required, let alone the three components together as one integrated continuously operating system.

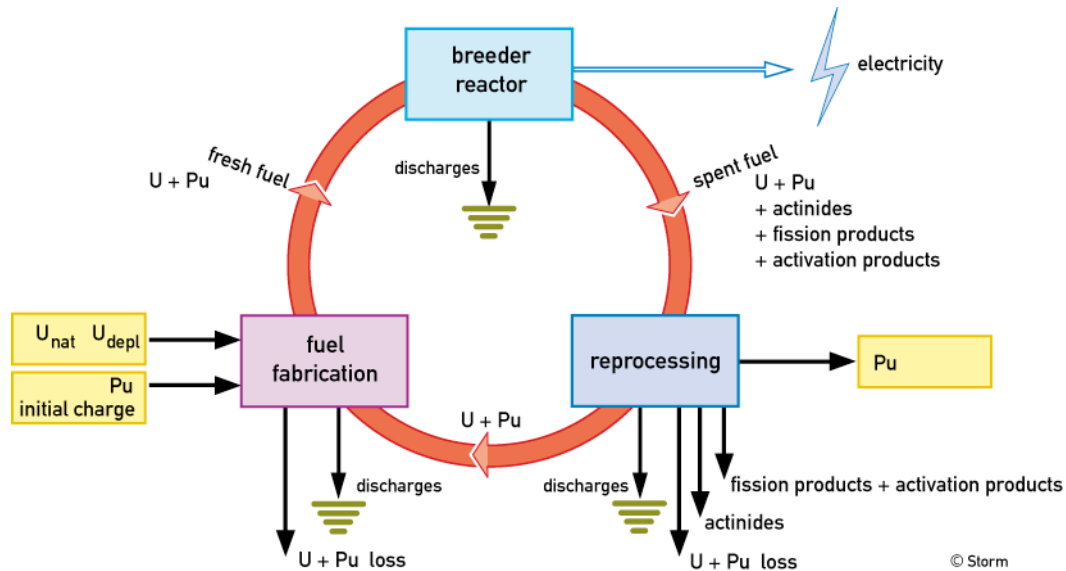


Figure B.3
 General outline of the breeder system in steady state. By repeatedly recycling spent fuel, it would be theoretically possible to fission the main part of natural uranium. If all goes well, the cycle produces during its operational life a plutonium gain, large enough to start up two or more new breeders: one to replace the closed down unit, and one or more additional breeders. The cycle represents the mass flows of uranium and the nuclides originating from the nuclear processes in the reactor (fission, activation and decay). The initial plutonium charge to start up the breeder reactor is about 3 Mg Pu for a 1 GW(e) FBR.

Fifty years of intensive research in seven countries (USA, UK, France, Germany, former USSR now Russia, Japan and India), with investments of a hundred of billions of dollars so far have failed to demonstrate that the breeder cycle is technically feasible. No technical breakthroughs are reported contradicting the unfeasibility of the breeder cycle.

Problems of the breeder system are discussed in more detail by, among others, UNIPEDE/CEC 1981 [Q58] and Lidsky & Miller 1998 [Q301]. The authors concluded that the breeder system is not feasible, not only due to the technical hurdles, but also because the system cannot meet the requirements of safety, proliferation and economy.

Thorium breeder

The thorium breeder is based on the conversion by neutron capture of non-fissile thorium-232 into fissile uranium-233, by a similar system as the uranium-plutonium breeder. The feasibility of the thorium breeder is even more remote than that of

the U-Pu breeder.

Problems include:

- the high radioactivity of U-233, which is always contaminated with traces of U-232,
- similar problems in recycling thorium due to the highly radioactive Th-228,
- technical problems not yet satisfactorily solved in reprocessing.

An overview of research projects in the past and of advanced thorium reactor concepts is given in [Q302].

Research and development on the thorium cycle has been less intensive than on the U-Pu cycle and never reached the prototype phase, like the U-Pu cycle with the French Superphénix. India still conducts some research on thorium-U-233 fuel cycle.

Besides, only minute quantities of U-233 exist in the world at this moment. It would take decades to obtain sufficient U-233 from special reactors to start up the first operating Th-232-U-233 breeder system. After that it would take 9 doubling times to attain a thorium breeder capacity equalling the current nuclear capacity (about 370 GW). Even with an unrealistically assumed short doubling time of 20 years that would mean two centuries.

Next decades

The MIT 2003 study *The Future of Nuclear Power* [Q280], does not expect breeders (in effect the breeder cycle) to come into operation during the next three decades. The MIT study concluded that for the next three decades, and probably beyond, nuclear energy generation has to rely on thermal-neutron reactors, mainly LWRs, in the once-through mode. The IAEA (Omoto 2007 [Q359]) does not expect the first fast reactor or breeder of Generation IV to come on line before 2040.

For that reason, combined with the considerations mentioned above, this study is based on the light water reactor LWR in the once-through mode (see also the decision tree in Figure B.1). The differences between the Boiling Water Reactor BWR, Pressurized Water Reactor PWR and the Russian VVER (also a PWR) with respect to the energy analysis are minor. The results of the energy analysis of the reference LWR can be considered to apply to all three light water reactor types.

In the once-through mode no uranium and plutonium are recycled, consequently spent fuel is not reprocessed. The authors of MIT considered the proliferation and safety risks of reprocessing and the use of mixed-oxide (MOX) fuel unjustified. But there are also economic reasons not to recycle in their view.

Studies by the Oxford Research Group show that MOX fuel poses a large and underrated terrorist risk (Barnaby 2005a, 2005b, 2006 [Q339], [Q340], [Q341], Barnaby & Kemp 2007 [Q360]).

For above reasons the LWR with plutonium recycling is not included in this study.

B2 Primary reactor parameters

The energy analysis of the nuclear system in this study [Q6] is based on a light-water reactor (LWR) of current design, in the once-through mode.

The parameters in Tables B.1 and B.2 are independent of the methodology and the assumptions of the energy analysis in this study.

Table B.1

Basic parameters of the reference reactor in this study. The mass quantities are explained in the uranium mass balance: see Figure B.4 and Table B.2.

Quantity	Symbol	Unit	Value
Power, electric (at grid connection)	P_e	GW	1.000
Power, thermal	P_{th}	GW	3.125
Nominal average burnup	B	GW.day/Mg	46
Tails assay of enrichment process	x_t	% U-235	0.20
Reload period	D	GW.year (= FPY)	0.82
Fraction of core replaced at reloading	-	-	0.25
Heat production per reload period	E_{th}	PJ/D	80.81
Heat production per reload period	E_{th}	10^9 kWh/D	22.45
Gross electricity production per reload period	E_e	PJ/D	25.86
Gross electricity production per reload period	E_e	10^9 kWh/D	7.183
First core			
Mass enriched uranium	m_0	Mg	81.20
Enrichment assay	x_p	% U-235	3.3
Enrichment feed/product ratio	F/P	-	6.08
Natural uranium feed	m_3	Mg	503.6
Front end feed ratio	m_3/m_0	-	6.20
Separative work	S	MSWU	0.4166
Reload charge			
Mass enriched uranium	m_0	Mg	20.30
Enrichment assay	x_p	% U-235	4.2
Enrichment feed/product ratio	F/P	-	7.84
Natural uranium feed	m_3	Mg	162.35
Front end feed ratio	m_3/m_0	-	8.00
Separative work	S	MSWU	0.1447

Uranium mass balance

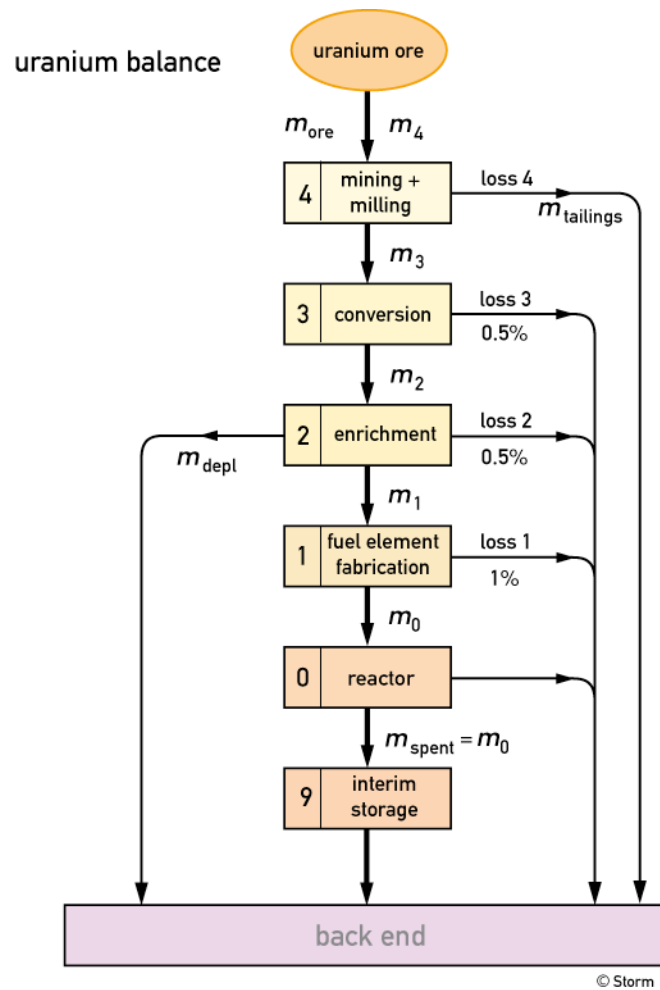


Figure B.4
Generic uranium mass balance of the LWR once-through system. The numbers of the processes refer to the scheme of the nuclear process chain as used in this study (see Part E.1).

Table B.2
Uranium mass balance (in Mg) of the LWR once-through chain

Process	Symbol	Relationship	Reload $x_p = 4.2\%$	First core $x_p = 3.3\%$
Conversion feed	m_3	$= 1.005 \cdot m_2$	162.35	503.6
Enrichment feed	m_2	$= F$	161.54	501.1
Feed/product ratio	F/P		7.84	6.08
Enrichment product	P	$= m_1 + \text{loss}2 = 1.015 \cdot m_0$	20.60	82.42
Depleted uranium	m_{depl}	$= F - P$	140.93	418.7
Fuel element fabrication	m_1	$= 1.01 \cdot m_0$	20.50	82.01
Reactor	m_0		20.30	81.20

Uranium first core and reloads

After each reload period one quarter of the fuel is removed from the core and replaced by an equal mass of fresh fuel, here called a reload charge. A *reload period* (symbol D) is the period during which a fixed amount of fissile nuclides is fissioned in the reactor and consequently a fixed amount of heat is generated (see Table B.1).

Due to operational conditions, the length in time of a reload period is variable.

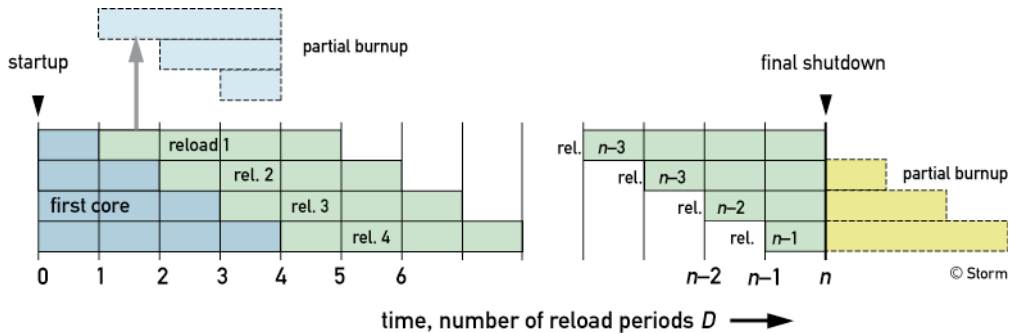


Figure B.5
Exchange schedule of nuclear fuel in the LWR once-through system

The reactor starts operation with a first core of fresh fuel. After the first reload period one quarter of the core is exchanged by a reload charge. The removed fuel has reached only a partly burnup, meaning that only a part of the fissile nuclides in the removed fuel are fissioned. After the second reload period a second quarter of the core is exchanged for fresh fuel, and so on. In this way $3/8$ of the uranium in the first core is removed unused (see Figure B.5).

At the end of its operational life the reactor is closed down. After a cooling period, the last core is removed from the reactor, again containing $3/8$ of the fuel unfissioned.

Summed up $3/4$ of the fissile content of the reactor core - an equivalent of 3 reload charges - leaves the nuclear system unfissioned during its operational life. This fact results in a higher total energy input of the system and a higher lifetime-averaged uranium consumption per unit delivered gross energy.

Reload period

As explained above the reload period of the reference reactor has a fixed value in terms of energy and nuclear fuel mass. This value is set by the primary reactor parameters (see Table B.1). Therefore the reload period is used in this study as the unit of operational lifetime.

The relationship between reload periods and operating years of the reactor is discussed in the next section.

B3 Load factor and operational lifetime

The operational lifetimes of the world nuclear reactors vary widely. As this study is aimed at the potential of nuclear energy for the world energy supply and mitigation of CO₂ emission, only world averages are used. The reference reactor is assumed representative of the future world average reactor. This may be an optimistic assumption, as the world average reactor today has a significantly lower performance.

Usually the lifetime of a reactor is given in years. From an energy point of view not the age in years of a nuclear power plant but its lifetime energy production is an important measure of performance. Here we introduce two performance parameters: the *load factor* and the *operational lifetime*.

Load factor

The reference reactor has a nominal power of 1 GW(e). The theoretical maximum amount of electricity the reactor can produce during one year, if the reactor would operate during that year continuously at 100% of its nominal power, is given by equation B.1.

$$\begin{aligned} J_{100} &= 1 \text{ GW}\cdot\text{year/a} = \\ &= 365 \cdot 24 \text{ GWh/a} = 8760 \text{ million kWh/a} = \\ &= 365 \cdot 24 \cdot 3600 \text{ GJ/a} = 31.536 \text{ PJ/a} \end{aligned} \quad \text{Eq B.1}$$

A reactor with a nominal power of P_o GW(e) has a maximum annual production of:

$$J_{100} = P_o \text{ GW}\cdot\text{year/a} = P_o \cdot 31.536 \text{ PJ/a} \quad \text{Eq B.2}$$

The load factor of the reactor during a given year i is here defined as the ratio of the actually produced amount of electricity during year i , $E_{\text{actual}}(i)$, and the theoretical maximum $J_{100}(i)$, (see equation B.1/2). Both quantities are taken as the gross values, measured at the connection of the nuclear power plant with the grid.

$$\text{load factor} = L_i = \frac{\text{actual gross energy output in year } i}{\text{maximum gross energy output in year } i} = \frac{E_{\text{actual}}(i)}{J_{100}(i)} \quad \text{Eq B.3}$$

The lifetime average load factor is the ratio of the actual lifetime electricity production, $E_{\text{actual}}(\text{life})$, and the theoretical maximum during that same period, $E_{100}(\text{life})$:

$$L_{\text{life}} = \frac{E_{\text{actual}}(\text{life})}{E_{100}(\text{life})} \quad \text{Eq B.4}$$

Operational lifetime

In this study we define the *operational lifetime* as the product of the age in years and the lifetime average load factor, according to equation B.5:

$$T_{100} = n_y \cdot L_{\text{life}} \quad \text{Eq B.5}$$

T_{100} = operational lifetime

n_y = numbers of years between startup and closedown

L_{life} = lifetime average load factor

The unit of T_{100} is years. In fact T_{100} is the number of full-power years (FPY): the virtual time of continuous operation at 100% of its nominal power in which the reactor would have generated an equal amount of electricity as it actually produced.

The operational lifetime T_{100} can also be calculated by equation B.6:

$$\text{operational lifetime} = T_{100} = \frac{E_{\text{actual(life)}}}{J_{100}} \quad \text{Eq B.6}$$

$E_{\text{actual(life)}}$ = actual lifetime gross electricity production (unit: J)

J_{100} = maximum annual electricity production (unit: J/a)

Historical data

Historical data (see Figure B.6) show that the average load factor L , as defined equation B.3, declines with age.

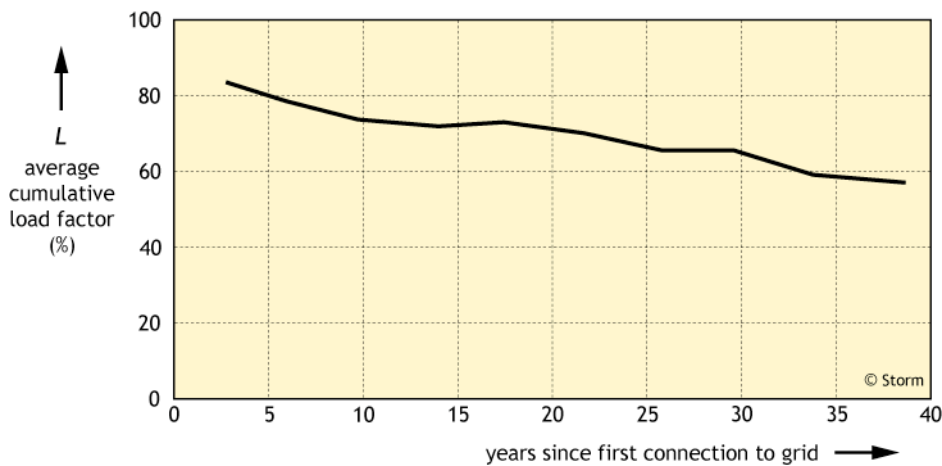


Figure B.6

Historical relationship between lifetime average load factor and the lifetime measured as the number of years connected to the grid. Five-year averages. This diagram includes data through 2001 (see text). Data from *Nuclear Engineering International*, November 2001, pp40-44.

The diagrams of Figures B.6 and B.7 have not been updated through 2006, due to a rise of inconsistencies in the statistical data after 2001 in the open literature. Recent data are becoming increasingly less comparable to older ones, owing to some statistical tricks practised by the nuclear industry.

1 Some nuclear power plants have officially derated their nominal capacity P_o . With the same annual electricity production, this results in a higher load factor.

2 A number of nuclear power plants have uprated their electricity production capacity, e.g. by operating at higher-than-design temperatures and pressures, without uprating their nominal capacity P_0 with an equivalent amount. This method has the same effect as trick 1: the load factor rises. In this way it became possible that plants reported load factors higher than 100%, see for example *Nuclear Engineering International*, May 2003, p 42.

3 Plants off-line during a year or longer, e.g. due a major overhaul or a long unplanned outage, are often deleted from the statistics of the average load factor of the concerned year. Consequently the average load factor of the remaining plants goes up.

4 Not all plant owners use exactly the same definition and/or name of the load factor L . Confusion may easily arise when other parameters are used, such as ‘capacity factor’ or ‘availability factor’, without explanation. So it may be conceivable that a reactor gets a rating of 100%, but did not operate actually during the period at issue, due to external factors.

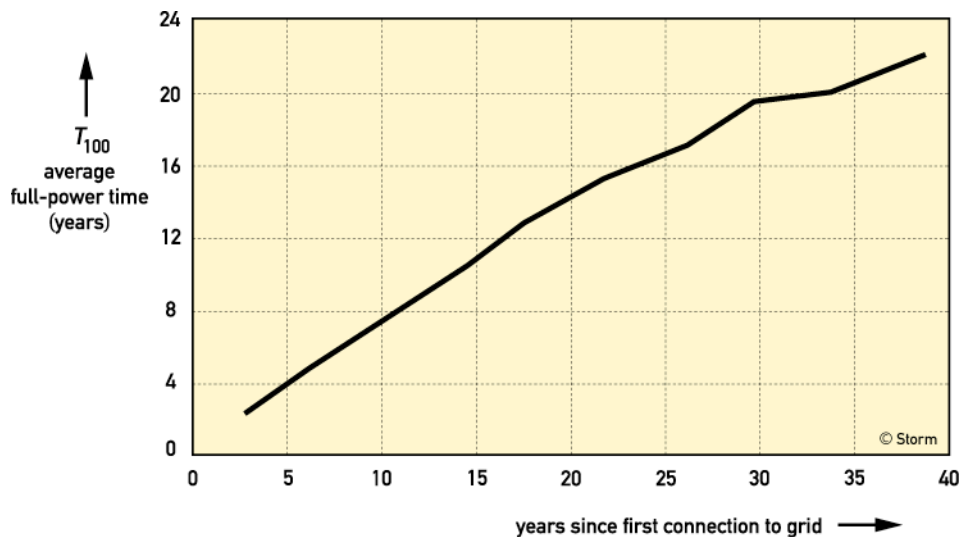


Figure B.7
Historical relationship between the full-power time and the lifetime measured as the number of years connected to the grid. Five-year averages. This diagram includes data through 2001 (see text). Based on data from *Nuclear Engineering International*, November 2001, pp40-44.

Reload period

The gross electricity production of the reference reactor per reload period D (see Figure B.5) has a fixed value:

$$J_D = 0.82 \cdot 31.536 = 25.86 \text{ PJ/D} \quad \text{Eq B.7}$$

The lifetime gross energy production of the nuclear power plant during n reload periods is:

$$E_{\text{grid}} = n \cdot J_D = n \cdot 25.86 \text{ PJ} \quad \text{Eq B.8}$$

The number of reload periods D can easily be converted into years, assuming any load factor L of the reactor, according equation B.9:

$$1 D = \frac{0.82}{L} \text{ years} \qquad \text{Eq B.9}$$

Operational lifetimes in this study

To illustrate the effect of the operational lifetime on the specific CO₂ emission and net energy production of nuclear power, we assessed three cases:

- 1 baseline (reference reactor in this study)
 $T_{100} = 30$ years at an average load factor of 0.82 = 30•0.82 = 24.6 full-power years (FPY) = 30 reload periods D
- 2 as in ISA 2006 [Q325]:
 $T_{100} = 35 \cdot 0.85 = 29.75$ FPY = 36.28 D
- 3 as, for example, in ExternE-UK 1998 [Q308] and Vattenfall 2005 [Q152]:
 $T_{100} = 40 \cdot 0.85 = 34.0$ FPY = 41.46 D

Outlook

By the end of June 2001 only four nuclear power plants in the world reached a full-power time of 24 years (FPY). The graph in Figure B.7 points to an average full-power time of about 22 FPY for plants aged 40 years.

The graph in Figure B.6 shows a decreasing tendency of the mean load factor with the age of the nuclear power plants..

In 2004 the mean age of the 440 nuclear power plants in the world was 21 years. The mean age of the 107 permanently shutdown NPPs of the world was 20.8 years in 2004 (Schneider & Froggatt 2004 [Q342]).

Whether nuclear power plants in the future will reach a world average of 24.6 FPY (30 years at a mean load factor of 0.82, the baseline case of this study) remains to be proven by empirical evidence. The existing evidence, as discussed above, points to a lower value. The higher values of 29.75 FPY or even 34 FPY seem remote.

B4 Secondary reactor parameters

The structure of this study is outlined in Figure B.8. The numerical results which are not dependent on the methodology and input data of the energy analysis are represented by the document icons at the left side of the diagram. The other parts of this study, which are dependent on input data and methodology, are on the right side of the diagram.

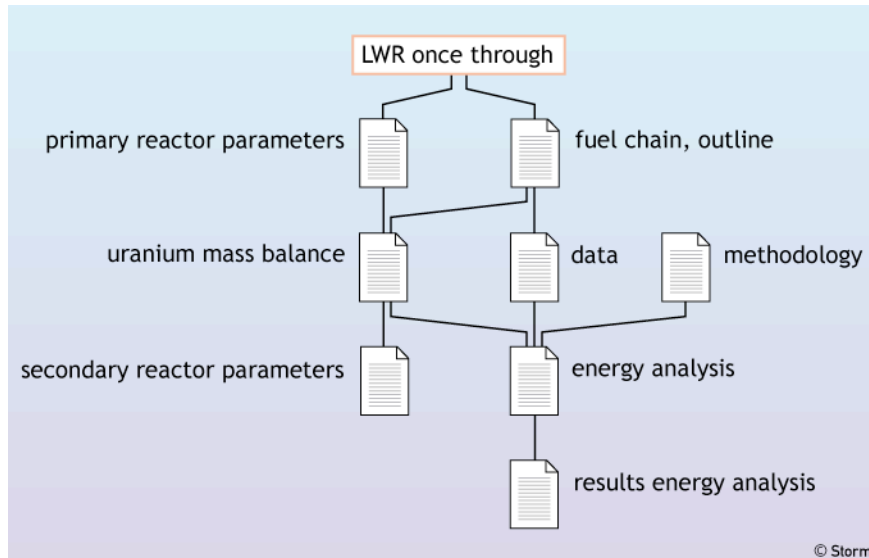


Figure B.8
Relationship between the main parts of this study

Lifetime-dependent reactor characteristics

The performance of the nuclear energy system depends, among several other variables, on the operational lifetime of the reactor. In other studies a fixed value of the lifetime has been chosen, in this study the operational lifetime is a variable and the reload period is used as the unit of the operational lifetime.

Lifetime uranium consumption

The uranium requirements during a life of n reload periods can be calculated by equations B.10 and B.11.

Natural uranium:

$$m_{\text{life}} = m_3(\text{fc}) + (n-1) \cdot m_3(\text{rel}) = 503.6 + (n-1) \cdot 162.35 \text{ Mg } U_{\text{nat}} \quad \text{Eq B.10}$$

Enriched uranium:

$$m_{\text{life}} = m_0(\text{fc}) + (n-1) \cdot m_0(\text{rel}) = 81.20 + (n-1) \cdot 20.30 \text{ Mg } U_{\text{enr}} \quad \text{Eq B.11}$$

Burnup

The burnup of nuclear fuel is defined as the heat production in the reactor per Mg enriched uranium. Usually the non-SI unit thermal gigawatt-year per Mg enriched uranium is used. The conversion factor c is:

$$c = 1 \cdot 3600 \cdot 24 = 86.400 \text{ TJ/ GW(th).day} \quad \text{Eq B.12}$$

The burnup, B , of a given quantity of nuclear fuel (enriched uranium), m , after producing E TJ thermal energy, is given by equation B.13:

$$B = \frac{E_{\text{th}}}{m_{\text{U(enr)}} \cdot c} \quad \text{GW.year/Mg} \quad \text{Eq B.13}$$

The heat production of the reference reactor per reload period (see Table B.1) is:

$$E_{\text{th}} = 80.811 \text{ PJ/D} = 80811 \text{ TJ/D} \quad \text{Eq B.14}$$

This corresponds with a burnup of:

$$B = (80811/20.30)/86.40 = 46.07 \text{ GW(th).day/Mg} \quad \text{Eq B.15}$$

This agrees with the nominal average burnup of the reference reactor: $B = 46$ GW(th).day/Mg in Table B.1.

The effective average burnup achieved by the reference reactor depends on the lifetime of the reactor and can be calculated by equation B.16:

$$\begin{aligned} B &= \frac{E_{\text{th}}}{m_{\text{U(enr)}} \cdot c} = \frac{n \cdot 80811}{m_{\text{U(enr-life)}} \cdot c} \\ &= \frac{n \cdot 80811}{(81.20 + (n-1) \cdot 20.30) \cdot 86400} \quad \text{GW.year/Mg} \end{aligned} \quad \text{Eq B.16}$$

Table B.3 gives the values of the effective burnup of the reference reactor at three different lifetimes.

Table B.3

Derived parameters of the reference reactor. The masses of consumed natural uranium and enriched uranium are calculated according equations B.10 and B.11 respectively.

Quantity (lifetime)	unit	Operational lifetime = years x average load factor		
		30x0.82	35x0.85	40x0.85
n = number of reload periods D	-	30	36.28	41.46
Total mass natural uranium	Mg	5212	6231	7073
Average mass U_{natural} per reload	Mg/D	173.74	171.76	170.59
Average mass U_{enriched} per reload	Mg/D	22.33	21.98	21.77
Heat production	PJ	2424.3	2931.8	3350.7
Heat production	10^9 kWh	673.5	814.4	930.7
Heat production, per Mg U_{nat}	TJ/Mg	465.2	470.5	473.7
Heat production, per Mg U_{enriched}	TJ/Mg	3618.9	3676.7	3711.9
Gross electricity production	PJ	775.8	938.2	1072.2
Gross electricity production	10^9 kWh	215.5	260.6	297.8
Gross electric. prod. per Mg U_{nat}	TJ/Mg	148.9	150.6	151.6
Gross electric. prod. per Mg U_{nat}	MWh/Mg	41.35	41.82	42.10
Natural uranium consumption	g/kWh(e)	0.0242	0.0239	0.0238
Total mass enriched uranium	Mg	669.90	797.39	902.61
Effective average burnup (1)	GW.day/Mg	41.89	42.55	42.97
U-235 utilization ratio (2)	-	0.74061	0.74058	0.74036
Fissioned fraction (3)	-	0.00574	0.00580	0.00584

(1) See equation B.16

(2) See equation B.19

(2) See equation B.22

Uranium-235 utilization ratio

The uranium-235 utilization is the ratio of the mass of the uranium-235 entering the reactor and the mass of the uranium-235 in the natural uranium leaving the mine.

The U-235 content of natural uranium is 0.71%, The mass of the natural uranium consumed during the reactors lifetime is given by equation B.6. The U-235 content of that quantity of natural uranium is:

$$m_3(\text{U-235}) = 0.0071 \cdot \{503.6 + (n-1) \cdot 162.35\} \text{ Mg} \quad \text{Eq B.17}$$

The enrichment assays of the enriched uranium placed into the reactor are:

first core: $x_p = 3.3\%$ U-235

reload charges $x_p = 3.3\%$ U-235 (see Table B.1).

The U-235 content of the enriched uranium entering the reactor is given by equation B.18.

$$\begin{aligned} m_0(\text{U-235}) &= x_p \cdot m_0(\text{fc}) + x_p \cdot (n-1) \cdot m_0(\text{rel}) = \\ &= 0.033 \cdot 81.20 + 0.042 \cdot (n-1) \cdot 20.30 \text{ Mg} \end{aligned} \quad \text{Eq B.18}$$

The uranium-235 utilization ratio R_{U235} is:

$$R_{U235} = \frac{\text{mass U-235 in reactor}}{\text{mass U-235 leaving mine}} = \frac{m_0(\text{U-235})}{m_3(\text{U-235})} =$$

$$= \frac{0.033 \cdot 81.20 + 0.042 \cdot (n-1) \cdot 20.30}{0.0071 \cdot \{503.6 + (n-1) \cdot 162.35\}}$$

Eq B.19

As equation B.19 shows, the U-235 utilization ratio is a function of the operational lifetime. In Table B.3 the values at three lifetimes are given.

Fissioned fraction

The fissioned fraction is here defined as the fraction of the natural uranium leaving the mine actually is fissioned in the reactor.

The average specific heat generation by fission of fissile nuclides is:

$$J_{\text{fission}} = 81.08 \text{ GJ/g} = 81.08 \text{ PJ/Mg}$$

Eq B.20

The total mass of the nuclides fissioned during the lifetime of the reactor can be found by equation B.21:

$$m_{\text{fission}} = (\text{gross lifetime heat generation}) / J_{\text{fission}}$$

Eq B.21

The lifetime fissioned fraction f is given by equation B.22:

$$f = \frac{n \cdot 80.81}{81.08 \cdot \{503.6 + (n-1) \cdot 162.35\}}$$

Eq B.22

As equation B.22 shows, the fissioned fraction is also a function of the operational lifetime. In Table B.3 the values at three lifetimes are given.

B5 The nuclear system

A nuclear reactor is not a stand-alone system. To make available the nuclear energy embodied in uranium, as found in the earth's crust, in the form of freely usable energy, numerous industrial processes are required. This complex of processes here is called the nuclear system. For practical reasons with regard to the energy analysis the nuclear system can be divided into two major chains:

- the construction-operation-decommissioning chain (represented by the blue arrows in Figure B.9)
- the nuclear fuel chain: from uranium ore to nuclear fuel to final sequestration of the radioactive waste (represented by red arrows in Figure B.9).

The fuel chain can be divided into three main parts:

- Upstream part (front end): conversion of uranium ore in the ground into fuel elements for the reactor,
- Reactor operation, maintenance and refurbishment of the nuclear power plant during its operational life span,
- Downstream part (back end): handling of the radioactive waste, decommissioning and dismantling of the (radioactive) nuclear reactor and sequestration of the waste in a safe geological repository.

Each of the main parts of the nuclear system comprises a number of industrial processes. Each process consumes electricity, fossil fuels, materials and chemicals, and emits carbon dioxide and some also other greenhouse gases. The operation of the reactor itself is the *only* part of the process chain which produces virtually no CO₂.

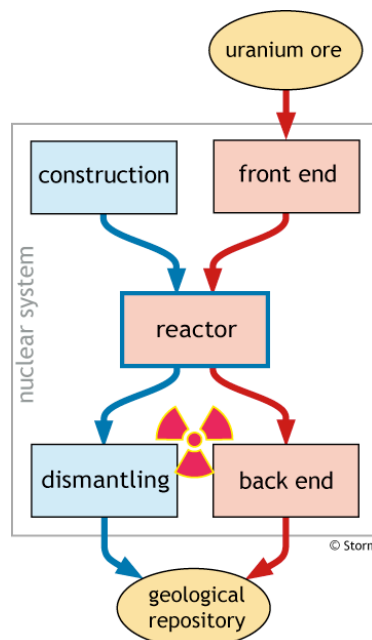


Figure B.9
General outline of the nuclear system.

the numerical results of the energy analysis. Other studies seem to circumvent this problem simply by deleting such processes from the analysis.

A full description of the processes of the nuclear chain can be found in the previous version of this study (August 2005) on the website of [Q6].

Focus

A major determinant of the future performance of the nuclear system, with respect to energy security and GHG emissions, turns out to be the thermodynamic quality of the uranium resources feeding the nuclear system. The thermodynamic quality of a uranium resource here is defined as the quantity of useful energy required to extract one kilogram of pure uranium from that resource. The main determinant of the thermodynamic quality is the grade of the uranium ore.

To assess the outlook of nuclear power, the thermodynamic quality of uranium resources in relation with the quantities of the known resources and yet-to-be discovered resources have to be assessed. Therefore this paper focuses on the energy input of the front end of the nuclear chain, in particular the recovery of uranium from its ore.

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