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Global Plutonium Production Capabilities with Civilian Research Reactors

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This article deals with the plutonium production capabilities in civilian research reactors and the resulting proliferation risks. A complete record of all civilian research reactors located in Non-Nuclear Weapon States and de-facto Nuclear Weapon States is compiled and systematized according to their type. A discussion of the various production paths and scenarios for plutonium with those reactors follows. In order to derive an assessment with a broad coverage, partly diverse fleet of reactors, two designs representative of light water and heavy water moderated reactors (which account for 82 percent of the total installed capacity in Non-Nuclear Weapon States and de-facto Nuclear Weapon States) were chosen and included in burn-up calculations with the Monte Carlo code KENO V.a and Origen-S, both incorporated in the modular system Scale-6. The effective production rates in fuel elements as well as by irradiation of targets are then applied to calculate the capabilities of plutonium production of each considered research reactor. The results provide an overview of the proliferation relevance of the global research reactor fleet and its regional distribution.

INTRODUCTION

Plutonium production in nuclear reactors is one of the main proliferation risks associated with the civilian application of nuclear energy. In the commercial energy sector, the storage and transportation of fissile material, especially of separated plutonium, are of major concern. The global stocks of separated plutonium today are in the range of hundreds of tons. On the other hand, historical experience shows that states with a clandestine military nuclear program have used small research reactors for the production of fissile material rather than diverting plutonium from a civilian nuclear power reactor program. For instance, Israel used a heavy water reactor (HWR) Dimona while India, the

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two HWRs CIRUS and DHRUVA; Pakistan has three HWRs at Khushab (plus one reactor under construction), and North Korea used a graphite-moderated reactor at Yongbyon.[1] States that reportedly pursued an unsuccessful nuclear weapon program in the past like Libya, Brazil, South Africa, and Iraq also had no operational nuclear power plants at the time of their programs (they chose highly enriched uranium [HEU] rather than plutonium as fissile material). This observation can be explained in several ways. First, states interested in a nuclear weapon program may simply not have power reactors, but only research reactors available for fissile material production. If they have any, the plutonium vector from spent fuel with high burn-up (similar to power reactors) is not favorable for nuclear weapons because of its relatively low share of fissile isotopes (plutonium-239 and plutonium-241) and its higher heat generation and radioactivity. Finally, the fuel or targets in research reactors are more easily accessible—and harder to safeguard by IAEA—than in power reactors. As such, a research reactor is the first choice if a state wants to conduct and pursue a military production program. The worldwide plutonium production capabilities in research reactors are significant when addressing proliferation issues.

This work reviews the worldwide research reactor fleet and conceivable plutonium production scenarios. For the most important types, burnup calculations are performed. The outcomes are effective production rates, which can be applied to derive quantitative conclusions on most of the civilian research reactors in Non-Nuclear Weapon States (NNWS) and de-facto Nuclear Weapon States (NWS).[2] The report theorizes the actually (inevitably) occurring plutonium production in the fuel as well as on the theoretical maximum capability to produce weapons-grade material. In doing so, the question of plutonium production capabilities of states which are not part of Annex II of the Comprehensive Nuclear Test Ban-Treaty (CTBT) will also be addressed. These Annex II states are required to ratify the treaty before it can be entered into force. Considerable production capabilities of states not part of the Annex II would undermine the intent of the current conditions for the entry into force of the treaty and will consequently have a high impact on the current nonproliferation regime politically.

Research Reactors and Their Plutonium Production Capability

Research reactors are used for a variety of purposes all over the world. They provide neutrons for applied or basic research in natural sciences and for studying radiation effects. Furthermore, special isotopes for medical or industrial purposes are produced in some of these reactors. Finally, they are often used for training and teaching purposes. Research reactors are used by a great variety of states; many of them do not have a commercial nuclear power sector. The Research Reactor Database (RRDB) of the IAEA provides a comprehensive

Table 1: Worldwide number of research reactors according to their status as provided by the IAEA Research Reactor Database as of 1 June 2011

Number of reactor units	Status
241	Operational
3	Under construction
2	Planned
13	Temporary shutdown
203	Shut down, but not yet decommissioned

overview of the world's research reactor fleet as well as critical and subcritical assemblies, including information on their basic technical specifications.[3] Table 1 provides an overview of the worldwide research reactor fleet and their operational status as of 1 June 2011. Altogether, 254 units are operational or in temporary shutdown, while five reactors are under construction or under planning. The thermal capacity of these units reaches from zero for subcritical/critical assemblies to 250 MW for the Advanced Test Reactor in the United States, but more than half of the units have a thermal capacity below 1 MW. The distribution of operational reactor units or assemblies according to the histogram of their thermal capacity is shown in Figure 1.

The plutonium production capability of a reactor is highly dependent on its thermal design capacity. The number of reactors to be analyzed can be reduced significantly by introducing a lower bound for thermal capacity under which plutonium production is negligible. With regard to plutonium production, safeguards of the IAEA concentrate on research reactors with a thermal capacity above 25 MW.[4] Although smaller reactors are also monitored by the IAEA, additional safeguards (more frequent inspections etc.) are applied for these large reactors to increase detection probability in case of clandestine plutonium or uranium-233 production (uranium-233 is a fissile isotope which can be used for a nuclear weapon. It is produced by neutron capture of thorium-232). It is argued that this threshold is defined by the fact that below 25 MW, it is impossible to produce one significant quantity (SQ) per year by irradiation of fertile targets. However, for the subsequent analysis presented in this work, a lower threshold should be defined in order to achieve coverage of a broader variety of reactors in the first instance. Thus, the scenario in which a proliferator uses more than one reactor to produce plutonium or indulges in production for more than one year is included. The threshold for this analysis is set to a thermal capacity (P_{th}) of 1 MW, which excludes 164 reactors. From the five NWSs, there are 41 research reactors above the defined threshold, two of which are under construction and two under temporary shutdown. These reactors are also not considered further because they are not relevant with regard to horizontal proliferation. Since it is the aim of this work to assess production capabilities of

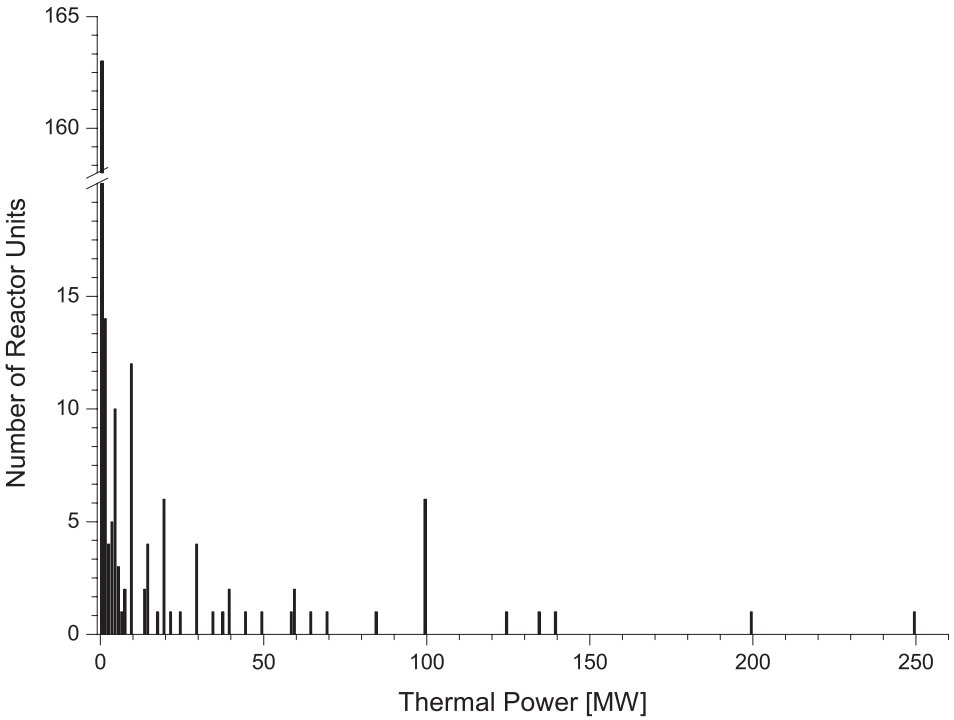

Figure 1: Number of operational research reactors in the world clustered by thermal capacity. The size of one class is 1 MW. Data is taken from IAEA's Research Reactor Database as of 1 June 2011.[41]

civilian reactors, the dedicated military production reactors of Pakistan (three operational reactors at Khushab and one under construction) and Israel (one reactor at Dimona) are not considered. The residual number of reactors (a total of 54 reactors in NNWSs and de-facto NWSs) is more manageable for the scope of this work. On the other hand, the defined threshold is certainly not too high to disregard significant plutonium production capabilities. Assuming a theoretical upper production rate limit of 0.94 g Pu/MWd as calculated by Binford,[5] the time required to produce one SQ of plutonium (8 kg) with a 1 MW reactor is about 23.2 years (with a load factor of 100 percent). In reality, this rate will be significantly lower due to leakage, other fissions than uranium-235, and so on. Therefore, the exclusion of reactors with a thermal energy below 1 MW in this analysis is rather conservative.

The IAEA generally distinguishes between three scenarios for the clandestine production and diversion of fissile material for military purposes.[6] These include the diversion of fresh fuel or slightly irradiated fuel, the diversion of spent fuel, and the clandestine production of plutonium or uranium-233. The first scenario is not considered in this work because it normally does not primarily address plutonium but rather only negligible amounts (apart from

exceptions like experiments including mixed-oxide fuel). Furthermore, the fresh fuel especially interesting for theft or diversion is HEU (direct-use material) but this is relatively unsuitable for plutonium production because of low concentrations of fertile nuclides. As such, spent fuel diversion and clandestine plutonium production can be addressed, in principle, by the distinction of three operational modes.

Normal Operation—Scenario A

The reactor is used for research purposes or the production of medical isotopes. In this case, the end of the cycle is reached when criticality drops below $k_{eff} = 1$ exploit the fuel as long as possible and thus keep the fuel costs low. Plutonium is inevitably produced in the reactor core. The plutonium vector of the discharged fuel is probably unfavorable for a weapon because of high burn-ups, but the material remains relevant to proliferation because it is still technically possible to build a nuclear weapon. After the fuel is discharged it will be stored but, in principle, still available for a proliferator. After a certain cooling period (several months at minimum), the fuel could be reprocessed. As spent fuel is subject to IAEA safeguards, it is routinely sealed and periodically controlled. Thus, it would be difficult to hide any diversion of plutonium. Furthermore, the diversion and reprocessing of spent fuel with high burn-up is technically more demanding than the plutonium extraction from targets because of the higher level of radioactivity and heat generated by the fuel. Still, a non-negligible risk of diversion might exist and, moreover, this scenario is important to consider for the case of open withdrawal of a state from the nonproliferation regime.

Optimization for Irradiation of Targets—Scenario B

Certain positions in the reactor core or in its close vicinity could be used for placing fertile targets with high concentrations of uranium-238. For example, irradiation channels which are normally present for research purposes can be used for undeclared irradiation.[7] The volume available for targets is, of course, limited by the reactor core design. Natural uranium (NU) or depleted uranium (DU), in the form of oxide or metal, would probably be used as target material. The purposes of research reactors require a core design that is in fact relatively easy to access. Insertion and retrieval of irradiation targets for research purposes or civilian isotope production are frequent activities in many facilities. This makes safeguards more difficult and costly. The actual accessibility of a core depends on the reactor type. For example, it is easier to remove fuel elements or targets from an open pool arrangement than from tank type research reactors. Generally, this scenario provides the highest probability of successful undetected diversion of plutonium out of the three considered here.

Full Core for Weapon Material Production—Scenario C

The whole reactor core could be used for weapons material production. This scenario is relevant if there is no distinction between driver fuel and target material, e.g., in case of HWRs that are completely fuelled with NU. The discharge burn-up is not predetermined by criticality, but by the isotopic composition of the plutonium produced in the fuel assemblies. The maximum burn-up is then marked by the limit of 7 percent plutonium-240 to ensure the material is weapons-grade.[8] Although it is a somewhat artificial limit because fuel grade or reactor grade plutonium can also be used for an explosive device,[9] this threshold assumes that the proliferator is interested in material which is most effective for military purposes.

All scenarios represent ideal cases. A real proliferator would perhaps choose modified fuel cycle patterns to avoid safeguards detection or accept plutonium with a more unfavorable isotopic composition. But in the course of this study, the scenarios seem to be reasonable because they cover the most important diversion pathways and particularly allow the estimation of upper production limits. However, if the IAEA is able to apply all necessary safeguards to a research reactor and its fuel, a clandestine diversion or production of fissile material in any of these scenarios will be quite difficult (with the highest chances of success for scenario B). This analysis is more relevant for a so-called break-out scenario in which a state pursues a military program without any intention of hiding it from the international community. Another conceivable possibility to decrease chances of detection while staying inside the non-proliferation regime could be the secret utilization of a reactor which is declared shut down.

Research Reactors in Non-Nuclear Weapon States

There are a total of 54 reactors in Non-Nuclear and de-facto NWS having a thermal capacity of above 1 MW which translates to 20 percent of the overall number of research reactors in the world. Out of them, five reactors are temporarily shut down and two are under construction. Table 2 provides an overview of these 54 reactors located in 36 countries as well as their technical specifications (thermal power, moderator, fuel type, enrichment). All reactors listed there will be included in this assessment of plutonium production capabilities in civilian research reactors. Both Indian heavy-water moderated reactors CIRUS and DHRUVA are dual-purpose plants, contained in the RRDB and therefore included in Table 2 (CIRUS has been shut down at the end of 2010,[10] but is still included in the list as it is useful in the assessment of the country's production capability which is the aim of this study).

Neutronics calculations for every single reactor would exceed the scope of this analysis by far. An assessment of plutonium production capabilities can only be made from classes of reactors with similar properties. Table 3 provides

Table 2: Research reactors in NNWSs and de-facto NWSs with P_{th} greater than 1MW according to IAEA RRDB. If LEU is specified as enrichment, no exact value could be found in the literature, but only the information that the reactor has been converted to an enrichment smaller than 20 percent. The only reactors not under safeguards are the Indian units and the one in the Democratic People's Republic of Korea.

Country	Facility name	P_{th} (MW)	Moderator	Fuel type	Enrichment	CTBT Annex II-status
Algeria	ES-SALAM	15	Heavy water	Unknown	LEU	yes (R)
Argentina	RA-3	10	Light water	MTR	LEU	yes (R)
Australia	OPAL	20	Light water	MTR	LEU	yes (R)
Bangladesh	TRIGA MARK II	3	Light water	TRIGA	19.75%	yes (R)
Belgium	BR-1	4	Graphite	Graphite	NU	yes (R)
	BR-2	100	Light water	Tank	93%	
Brazil	IEA-R1	5	Light water	MTR	19.75%	yes (R)
Bulgaria	IRT-SOFIA*	2	Light water	IRT-4M	19.75%	yes (R)
Canada	NRU	135	Heavy water	Other	20%	yes (R)
	MNR MCMASTER	3	Light water	MTR	19.75%	
Chile	RECH-1	5	Light water	MTR	19.75%	yes (R)
	RECH-2*	2	Light water	MTR	19.75%	
Czech Republic	LVR-15 REZ	10	Light water	IRT-2M	36%	no (R)
Egypt	ETRR-1	2	Light water	EK-10	19.75%	yes (S)
	ETRR-2	22	Light water	MTR	19.75%	
Germany	BER-II	10	Light water	MTR	LEU	yes (R)
	FRM II	20	Heavy water	Compact core	93%	
Greece	GRR-1*	5	Light water	MTR	19.75%	no (R)
Hungary	BUDAPEST RR	10	Light water	VVR-M2	LEU	yes (R)
India	CIRUS*	40	Heavy water	Rod	NU	yes (-)
	DHRUVA	100	Heavy water	Unknown	NU	
	FBTR	40	Fast breeder reactor			
Indonesia	BANDUNG	2	Light water	TRIGA	8.5%–20%	yes (S)
	GA SIWABESSY	30	Light water	MTR	19.75%	
Iran, Islamic Republic of Israel	TRR	5	Light water	MTR	19.75%	yes (S
	IR-40+	40	Heavy water	RBMK	NU	
	IRR-1	5	Light water	MTR	93%	yes (S)

(Continued on next page)

Table 2: Research reactors in NNWSs and de-facto NWSs with P_{th} greater than 1MW according to IAEA RRDB. If LEU is specified as enrichment, no exact value could be found in the literature, but only the information that the reactor has been converted to an enrichment smaller than 20 percent. The only reactors not under safeguards are the Indian units and the one in the Democratic People's Republic of Korea. *(Continued)*

Country	Facility name	P_{th} (MW)	Moderator	Fuel type	Enrichment	CTBT Annex II-status
Japan	JRR-3M	20	Light water	MTR	20%	yes (R)
	KUR*	5	Light water	MTR	20%	
	JRR-4	3.5	Light water	MTR	20%	
	JMTR	50	Light water	MTR	20%	
	JOYO*	140	Fast breeder reactor			
	HTTR	30	High temperature test reactor			
Jordan	JRTR+	5	Light water	MTR	19.75%	no (R)
Kazakhstan	WWR-K ALMA ATA	6	Light water	VVR-K	36%	no (R)
	EWG 1	35	Light water	Unknown	90%	
Korea, Dem. P. R. of	IRT-DPRK	8	Light water	IRT	unknown	yes (-)
Korea, Republic of	HANARO	30	Heavy water	Rod	19.75%	yes (R)
Libyan Arab Jamahiriya	IRT-1	10	Light water	IRT-4M	19.7%	no (R)
Morocco	MA-R1	2	Light water	TRIGA	19.7%	no (R)
Netherlands	HOR	2	Light water	MTR	19.75%	yes (R)
	HFR	45	Light water	MTR	19.75%	
Norway	HBWR	20	Heavy water	Rod	6%	yes (R)
	JEEP II	2	Heavy water	Rod	3.5%	
Pakistan	PARR-1	10	Light water	MTR	20%	yes (-)
Peru	RP-10	10	Light water	MTR	19.75%	yes (R)
Poland	MARIA	30	Light water	MR-6	19.75%	yes (R)
Romania	PITESTI	14	Light water	TRIGA	19.75%	yes (R)
South Africa	SAFARI-1	20	Light water	MTR	19.75%	yes (R)
Taiwan	THOR	2	Light water	TRIGA	20%	China: yes (S)
Thailand	TRR-1/M1	2	Light water	TRIGA	20%	no (S)
Ukraine	WWR-M KIEV	10	Light water	WWR-M2	19.75%	yes (R)
Uzbekistan	TASHKENT	10	Light water	IRT-4M	19.75%	no (R)
Total	54 units	1166.5				yes: 27, no: 9

Table 3: Research reactor types according to the IAEA's Research Reactor Database in NNWS and de-facto NWS with P greater than 1MW clustered according to their type

Reactor type	P_{th} (MW)	No. of units	No. of states	Moderator	Occurring enrichments	Coolant
Heavy water	402	9	7	D_2O	NU (3), LEU (1), 3.5% (1), 19.75% (1), 6% (1), 20% (1), 93% (1)	D_2O or H_2O
MTR	362.5	23	18	H_2O	LEU (3), 19.75% (12), 20% (5), 93% (2)	H_2O
Fast breeder	180	2	2	none	MOX	liquid Na
IRT (IRT-4M/2M)	32	4	4	H_2O	19.75% (3), 36% (1)	H_2O
HTR	30	1	1	Graphite	LEU	He
VVR (VVR-M2, VVR-K, EK-10)	28	4	4	H_2O	LEU (1), 19.75% (2), 36% (1)	H_2O
TRIGA	25	6	6	H_2O, ZrH	19.75% (3), 20% (3),	H_2O
Graphite	4	1	1	Graphite	NU	Air
Other LWR	103	4	4	H_2O	19.75% (2), 90% (1), unknown (1)	H_2O

Note: The thermal capacity refers to the sum of all reactors of the specified type. This list includes five reactors temporarily shut down, two reactors under construction, and one planned.

this classification on the basis of fuel geometry or, if no standard geometry can be identified, on the basis of used moderator. For every class, the installed capacity and fuel enrichments are stated. This list shows that HWRs and Material Test Reactors (MTR) are the dominant types, both in terms of thermal capacity and installed number. Other light water moderated designs are IRT, VVR and TRIGA. Fast Breeders constitute another major group on the basis of thermal capacity, but there are only two units present. Furthermore, there is one High Temperature Reactor and one Graphite Moderated Reactor. The fuel enrichments for LWRs vary from 19.75 percent to 93 percent. These enrichments, which are much higher than in power reactors, are used in research reactors because they are necessary to provide high specific reactivity allowing a compact core design with high power densities and thus high neutron fluxes. HWRs are partly fuelled with high enrichments, but can also be fuelled with NU. In comparison, NWSs used mostly NU fuelled reactors, either moderated with graphite (with water or gas as the coolant) or heavy water (with water or deuterium oxide as the coolant), to produce their weapons-grade plutonium. Few exceptions are only breeding targets in HEU-fuelled reactors. All four de-facto NWS also used NU fuelled reactors, mostly with heavy water as moderator.[11]

Methodology for the Assessment of Plutonium Production Capabilities

For the estimation of plutonium production in a nuclear reactor, burn-up calculations have to be performed, which take coupling of energy dependent neutron flux and the system of depletion equations for a specific geometry and materials into account. The result of the calculations is the plutonium content in fuel as a function of burn-up (in MWd/kg). The gradient of this curve is the amount of plutonium produced per MWd, or the production rate C. Since the gradient changes with burn-up (or with irradiation time, respectively) it has to be averaged over a certain burn-up interval. For calculating the plutonium produced in the reactor with a fixed discharge burn-up B_d, the averaged plutonium production rate M in [kg/a] is then given by

$$M = C_B \cdot L \cdot P_{th} \cdot 365\mathrm{d}$$

where C_B is the production rate averaged over $B = [0, B_d]$, L is the load factor and P_{th} the nominal thermal capacity. If one is interested in the annual production capacity at other burn-up intervals, C has to be evaluated over these intervals, whose length must correspond to one year of reactor operation.

SCALE-6 (Software Computer Analyses for Licensing Evaluation) is used to perform the burn-up calculations.[12] It is a computer code developed by Oak Ridge National Laboratory for the U.S. Nuclear Regulatory Commission and

consists of several modules that work together according to the control sequence called by the user. These modules allow neutronic transport calculations, criticality calculations, depletion calculations, and others. For burn-up calculations performed in this work, the control sequence TRITON-5 (Transport Rigor Implemented with Time-dependent Operation for Neutronic depletion) was used, which couples the three-dimensional Monte-Carlo transport code KENO V.a and the depletion code Origen-S.

An exact calculation of plutonium production would require modeling each reactor separately with neutronic transport codes taking all different fuel geometries, core dimensions etc. into account and exceeds the scope of this work. Therefore, the approach must require less simulation complexity but still be able to provide results applicable to a large majority of reactor units. For that purpose, two reactor designs, representative for a maximum of the reactors listed in Table 3, are identified and analyzed. The three other types listed in Table 1 are breeder reactors (2 units), high temperature gas cooled reactors (1 unit) and graphite moderated (1 unit). These reactors cannot be included here.

LWRs are the most common type of research reactors in the world. In 2010, they account for a total thermal capacity of about 550 MW (sum of types MTR, IRT, VVR, TRIGA and other LWR, without considering reactors with $P_{th} \leq$ 1MW). Their average thermal capacity is 14.1 MW per unit. As described in the previous section, the most common research reactor type that uses light water as moderator is the MTR and its plate-type fuel is the basis for calculating the plutonium production in light water research reactors.

Research reactors moderated with heavy water (and coolant) constitute the second largest group of research reactors worldwide. The sum of their thermal capacity equals 372 MW (including one reactor under construction). The average thermal capacity per unit is 46.5 MW, which is considerably larger than the average size of light water moderated units. Exemplary for this reactor type is that burn-up calculations for the Iranian HWR at Arak are performed for different production scenarios. Like all heavy water research reactors assessed here, it uses rod-type fuel assemblies. The following method determines whether Arak's reactor geometry is representative with regard to plutonium production in the other units. If moderator properties and fuel meat density are fixed, the conversion ratio C_R (number of fissile plutonium produced to the number of fissile uranium destroyed) of a reactor is primarily determined by the resonance escape probability p (simplified approximation provided by Bodansky[13]),

$$C_R = \frac{1-p}{\eta_0} + \frac{\eta_0}{\eta - 1},$$

with η as the number of fission neutrons per neutron absorbed in the fuel for a uranium isotope mixture and η_0 being the magnitude of η for pure uranium-235. Both η and η_0 only depend on fuel compositions. p is itself dependent on

the fuel geometry by the following approximate formula:

$$p = \exp\left[\frac{-N_F V_F I}{\zeta_M \Sigma_M V_M}\right],$$

with N_F being the atom density of fuel, $\zeta_M \Sigma_M$ moderator specific constants, I the resonance integral and V_M/V_F the volume ratio of moderator to fuel.[14] With given fuel meat density and moderator properties, p is only a function of the last two factors, where I possesses the following proportionality:

$$I \propto A + \frac{1}{\sqrt{r}}.$$

A is another constant and r is the radius of the fuel rod. Only V_M/V_F and r can be used for comparison of plutonium production in different reactors. Arak's fuel rods have a radius of r = 0.575 cm and a moderator-to-fuel ratio of $V_M/V_F = 31.4$ (lattice pitch of 25 cm assumed). The radius is quite typical for HWRs. CANDU fuel assemblies have a rod radius of 0.605 cm, the Canadian NRU reactor, a radius of 0.548 cm. Only the moderator-to-fuel ratio of Arak is a bit higher than that of other reactors (CANDU: 18.2, NRU: 21.3).[15] However, since research reactors, especially those moderated by heavy water, have no standardized design, it is generally very difficult to choose any representative model. But as shown, Arak has similar specifications concerning fuel geometry compared to other HWRs. In any case, the mode of operation (load factor, enrichment, etc.) has much more influence on the range of possible plutonium production capability than fuel geometry specifications. Given that the aim of this analysis is to give an approximate magnitude for plutonium production, choosing Arak as a model for other heavy water moderated reactors seems justifiable. Nevertheless, the relatively high moderator-to-fuel ratio of Arak could lead to an overestimation of plutonium production rates when extrapolating the results from Arak to other heavy reactors. Therefore, one should consider that choosing Arak is a rather conservative choice for calculating the world's plutonium production capabilities by heavy water research reactors.

An advantage of choosing Arak's geometry is that its exact plutonium production capability has not been calculated so far but is certainly of high interest, although no surprises regarding the order of magnitude are expected. Unfortunately, validity of the produced results for advanced fuels and reactor geometries is probably poor. Therefore, it will not be possible to assess a compact core design such as the German heavy water moderated reactor FRM II.

Plutonium Production Capability of Light Water Moderated Research Reactors

The design of MTR fuel was developed in the early 1950s by the United States and spread around the world in the following decades. It is a standard which has been adopted by several countries when they start nuclear their own research programs.[16] Today, there are 23 units with P_{th} greater than 1 MW in 18 of the analyzed countries (Table 3). The uranium (in the form of oxide or as alloy, for example with aluminum) in MTR is arranged in plates enclosed by aluminum. Several are combined to form a fuel assembly. The exact number varies from type to type and depends on whether it is a control assembly or a normal assembly (control assemblies have fewer plates in order to accommodate neutron absorbers). Typical values are 17–23 plates per assembly. The gap between the plates is filled with the moderator which functions simultaneously as coolant. Several of these assemblies are placed side by side in the rectangular reactor core surrounded by water or other reflectors like graphite or beryllium. Some fuel assembly positions may be available for the irradiation of targets.

For the subsequent calculation of plutonium production rates, a representative reactor configuration has to be chosen. In the context of core conversion programs, the IAEA proposed a generic 10 MW pool-type MTR for neutronic calculations,[17] which is used here. As the capacity is in the same range as the average existing capacity per reactor unit (14.1 MW), the choice is reasonable. The reactor core of IAEA's generic model consists of a 5×6 lattice with 23 standard (23 plates) and 5 control (17 plates) fuel assemblies, as well as two irradiation positions (one central and one edge). The core is reflected by graphite at the two short sides (the core as a whole consists, therefore, of a 7×6 lattice) and by water at the two other sides. The exact dimensions of the fuel assembly geometry are given in Table 4. The calculations are performed both for production scenarios relevant for light water reactors (scenarios A and B) for three different enrichments (19.75 percent, 36 percent, 93 percent), which are chosen in accordance with the values given in Table 3. The simulations with the Scale-6/Triton5 sequence were run for standard fuel elements in an infinite lattice for scenario A and for the whole reactor core configuration for scenario B.

The cycle length is limited by reactivity of the reactor fuel, frequency of maintenance, economic factors and the experimental program. Binford provides a typical value for cycle length equal to two to four weeks.[18] This also corresponds well to the length proposed for the generic MTR which is 16.7 d.[19] The subsequent refueling can take a few hours up to several days. The number of batches for the whole core is 7–9. The average discharge burn-up given by literature is consistently about 50 percent.[20, 21]

Table 4: Design parameters for the MTR generic 10 MW reactor core according to IAEA[1]

Item	Attribute	Description
Fuel Plate	Meat dimension	0.38 mm × 63 mm × 600 mm
	Meat composition	UAl-Al
		93% enrichment: 0.63 gU/cm^3, 21 weight-% U in UAl-Al
		36% enrichment: 2.50 gU/cm^3, 40 weight-% U in UAl-Al
		19.75% enrichment: 4.45 gU/cm^3, 72 weight-% U in UAl-Al
	Plate dimension	1.27 mm × 66.40 mm × 660 mm
	Cladding composition	aluminum, 2.7 g/cm^3
Assembly structure	Dimension	80.65 mm × 4.80 mm × 660 mm
	Composition	aluminum, 2.7 g/cm^3
Lattice pitch	-	77 mm × 81 mm
Moderator/coolant	-	H_2O

[1]International Atomic Energy Agency, 1980, *op. cit.*

Results for Production Scenario A

In this scenario, the standard MTR-fuel configuration without irradiation targets is used to calculate the amount of plutonium produced at different enrichment levels. Figure 2 shows the concentration of plutonium in the fuel meat for the three considered enrichments. The meat volume for one standard fuel element is 443 cm^3. HEU as fuel leads to negligible plutonium production (1.3 mg Pu/cm^3) at a burn-up of 50 percent due to the very low uranium-238 content in the fuel. The medium enriched fuel contains at this burn-up about 21.5 mg Pu/cm^3, whereas low enriched uranium (LEU) fuel reaches a concentration of about 35.0 mg Pu/cm^3. Figure 2 also shows the change of fissile fraction in the plutonium produced. The low and medium enriched fuel are very similar, whereas the 93 percent enriched fuel contains a progressively lower fissile fraction at higher burn-ups (77 percent at a burn-up of 50 percent compared to between 82.5 percent and 83 percent for the lower enriched fuels). Finally, Table 5 provides the plutonium production rates for the three enrichments and discharge burn-ups at 40 percent, 50 percent, and 60 percent (averaged over the whole burn-up period). It can be observed that a change of 50% ± 10%-points in burn-up leads to a change of plutonium production of four to ten percent, depending on the enrichment level. These factors can be applied for the plutonium production of research reactors in scenario A. The production rates derived here are in the same range as the values found in literature.[22] For example, a 10 MW reactor with a load factor of 75 percent fuelled with LEU (19.75 percent enriched) and a target-burn-up of 50 percent produces 260 g Pu/a with a fissile content of about 83 percent.

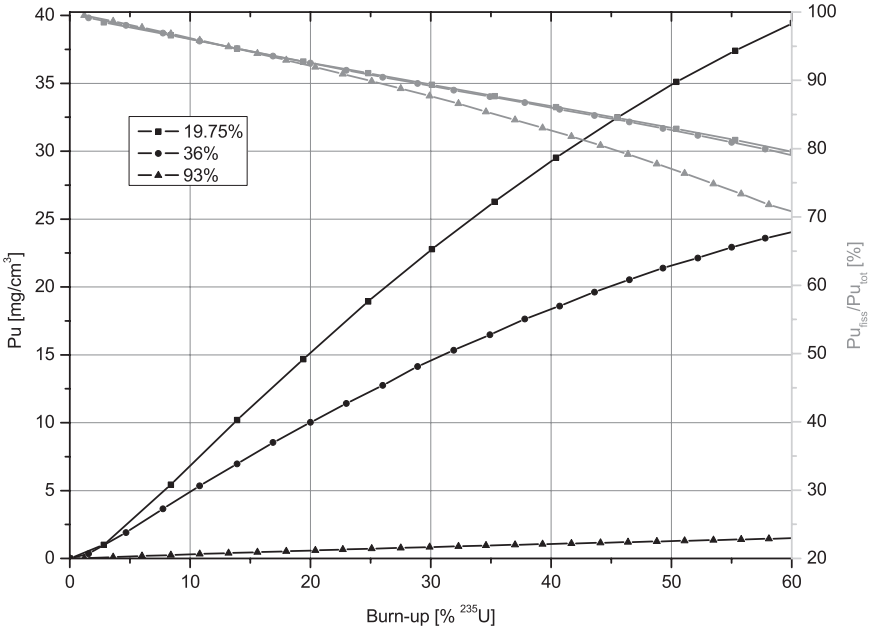

Figure 2: Concentration of plutonium as well as fissile plutonium fraction as function of burn-up for three enrichment levels for IAEA's generic MTR.

Results for Production Scenario B

In this scenario, it is assumed that the proliferator introduces targets with high amounts of uranium-238 into the core. This could be either NU (about 0.7 weight-percent uranium-235) or DU (about 0.3 weight-percent uranium-235). Concerning the plutonium production, both materials behave similarly under the same irradiation conditions.[23] Since NU is generally easier to produce or purchase than DU (the latter is a product of enrichment or reprocessing activities), these simulations assume NU as target material.

According to the MTR benchmark design, targets can be positioned at two points inside the core (one central position and one edge position) and at the

Table 5: Plutonium production rates for scenario A and with IAEA's generic MTR model for three initial enrichment levels

Enrichment (% uranium-235)	Production rate at 40% burn-up (mg Pu/MWd)	Production rate at 50% burn-up (mg Pu/MWd)	Production rate at 60% burn-up (mg Pu/MWd)
19.75	101	95.3	88.7
36	63.3	60.2	54.2
93	5.44	5.16	4.98

12 positions where the graphite reflector is normally located.[24] More targets could be positioned outside of the reactor's 7 × 6 grid but this would only be possible with significant design changes to provide sufficient cooling and is not considered here. Also not considered is the possibility of replacing fuel positions with irradiation positions. The target material is contained in MTR fuel plates that possess a larger fuel meat region than normal plates in order to accommodate high amounts of NU. Following the proposal of Miller and Eberhard,[25] 10 plates per target assembly are assumed, each with a meat thickness of 0.5 cm. Thus, a uranium load of 36.06 kg per target assembly in a volume of 1893 cm^3 can be achieved. If the target thickness is increased further, the self-shielding effects of uranium-238 will make the additional yield of plutonium production marginal.[26]

A proliferator has to consider three variables for the plutonium production for weapon's application. First is the quality of plutonium which is determined by the isotopic composition. With increasing irradiation time, the share of fissile plutonium decreases and so does its quality for weapons. Second is the magnitude of material input and associated workload for fuel fabrication, reactor operations, and fuel reprocessing. The highest production rates are achieved at low burn-ups, but then the fuel must be changed more often and more input material has to be processed per gram of plutonium output. This leads directly to the third variable, the productivity (relation between plutonium outputs in a certain period of time to the total production costs). The question of whether a proliferator has to act either more effectively or more efficiently depends on a variety of factors, which cannot be systematically addressed here. To get a reasonable upper boundary for plutonium production capability, it is assumed that the only leading constraint is the isotopic composition (weapons-grade plutonium).

Simulations were run with all three driver fuel enrichment levels and differentiated between central, edge, and reflector position. The concentrations given for the latter are values averaged over all twelve reflector positions. Figure 3 shows the results for burn-up calculations of the reactor configuration as described above, exemplary for a driver fuel enrichment of 19.75 percent. All graphs show linear behavior due to the generally low burn-ups. As expected, the central position shows the highest production rate due to the high neutron flux created by the surrounding driver fuel. The edge target position is still surrounded by driver fuel on two sides, and the reflector positions only on one side. This difference is reflected in the production rates, which are both considerably lower than the one in the central position. The influence of flux magnitudes produced by the different driver fuel can be identified in Figure 3 and Table 6, and illustrate the maximum amount of energy produced by the whole reactor core to assure weapons-grade material (greater than seven percent plutonium-240) in the targets. Table 6 also provides the corresponding production rates for plutonium. It should be noted that in this scenario, the

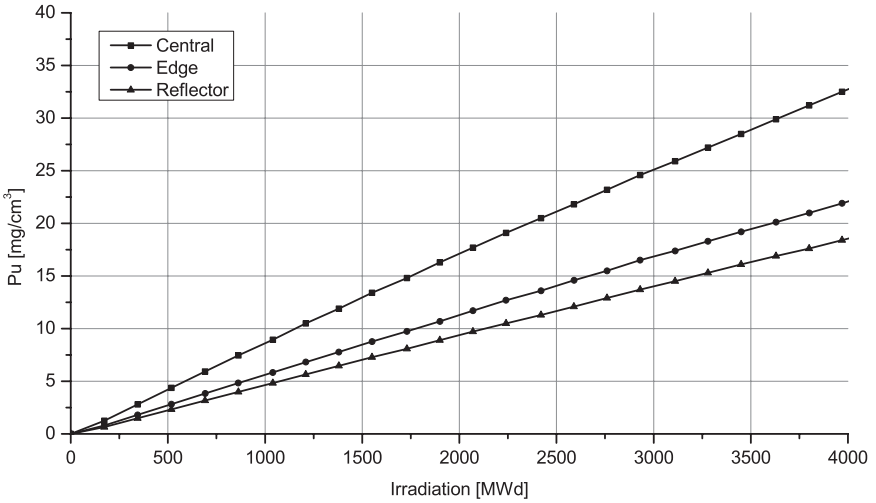

Figure 3: Plutonium production in NU-targets in central, edge, and reflector position (scenario B) for 19.75 percent enriched driver fuel. The concentration given for the reflector positions are averaged over all positions.

unit [mg/MWd] does not refer to the amount of energy output by the material in which the plutonium is produced (the targets), but to the reactor core as a whole.

Compared to Tomanin et al.,[27] the results indicate a much lower production capability, i.e., 0.44 gPu/MWd to 0.19 gPu/MWd (own calculation) for 93 percent enriched driver fuel. The difference is caused by the assumptions on how much target material could be introduced and irradiated in a research reactor. As described above, this quantity can vary significantly because

Table 6: Maximum energy produced by the core for a weapons-grade plutonium vector (greater than 7 percent plutonium-240) in the targets and corresponding plutonium production rates (scenario B) in IAEA's generic MTR model for three initial driver fuel enrichment levels

Enrichment (% uranium-235)	Energy produced until weapons-grade Pu vector in target (MWd)	Production rate central/edge/reflector (mg/MWd)	Total production rate for scenario B (mg/MWd)
19.75	4670	15.3 / 10.4 / 105	131
36	4650	15.7 / 10.7 / 108	134
93	3075	21.8 / 14.8/ 155	192

Note: The value for the energy production threshold in the second column is calculated with respect to the plutonium vector of the central target (the other targets reach the weapons-grade plutonium vector threshold a bit later). The rates refer to all targets assemblies in the particular positions (one central, one edge, and 12 reflectors).

single research reactor designs with same thermal power often differ in terms of grid size, number of irradiation channels, cooling capacity, and so on. If one assumes a core totally surrounded by targets, one could roughly achieve a total production capability of 0.45 gPu/MWd, which is close to the value of Tomanin et al. (10 additional reflector target positions using the production rate for 93 percent enriched fuel from Table 6).

Plutonium Production Capability of Heavy Water Moderated Research Reactors

In June 2004, Iran began to build a HWR (IR-40) near Arak, which has a thermal design capacity of 40 MW and is scheduled to be operational in 2013.[28] Estimations on the annual plutonium production capacity of a HWR with this size vary between 8 kg/a and 12 kg/a.[29, 30] Information about the final specifications of the Arak reactor is rare but some key information about the core and fuel geometry is published. Following these reports,[31, 32] the Arak reactor will be deployed with RBMK fuel assemblies which may contain NU or LEU. These fuel rod bundles are 3.5 m high and consist of 18 separate fuel pins, which are arranged in two concentric circles around a gas filled central tube. The bundle is encased by a pressure tube, giving the whole assembly an outer radius of 4.4 cm. Two bundles are placed vertically together forming a fuel cell with a height of 7 m. The clad materials of the pins and tubes are zirconium-niobium-alloys. Heavy water will be used for moderation and as a coolant. Detailed specifications from RBMK fuel assemblies, which are subsequently used for calculations, are provided with references in Table 7. The lattice pitch in RBMK reactors is 25 cm. If NU oxide is used, the total content of heavy metal in one fuel assembly is 122.4 kg. Since RBMKs use pressure tubes like CANDUs, it is likely that a reactor like Arak has the possibility of being refueled online.

Each calculation is performed for an infinite array of single fuel assemblies with 25 burn-up steps ranging from 0–30,000 MWd/kgHM for scenario A and 0–2,800 MWd/kgHM for scenario C. Besides NU fuel, three different enrichments are chosen in accordance with the values given in Table 3 for heavy water moderated reactors.

Results for Production Scenario A

Figure 4 shows the monotone decrease of the reactivity ρ, which is slow at very low burn-ups, faster at medium burn-ups, and decreases at high burn-ups, typical for NU-fuelled reactors. The high rate of fissile plutonium build-up at the beginning of the cycle contributes to reactivity resulting in slow decrease of ρ.[33] The initial fast drop of ρ between the first two data points is caused by build-up of fission products like xenon-135 during the first days of operation,

Table 7: Design parameters for Arak reactor core model based on the publications from Albright,[1,2] Murphy,[3] and the Nuclear Regulatory Commission[4]

Item	Attribute	Description
Fuel pellet	Radius	0.575 cm
	Composition	UO_2 (NU, 3.5%, 6% 19.75%)
Inner fuel bundle	Radius	1.605 cm
Outer fuel bundle	Radius	3.101 cm
Fuel pin	Outer radius	0.68 cm
	Inner radius	0.5975 cm
	Clad	Zr: 98.97%, Nb: 1%, Hf: 0.03%
	Fill gas	He (0.1 MPa, $5.36 \cdot 10^{-5} \frac{g^3}{cm}$)
Central tube	Outer radius	0.625 cm
	Inner radius	0.75 cm
	Clad	Zr: 97.47%, Nb: 2.5%, Hf: 0.03%
	Content	N_2
Assembly pressure tube	Outer radius	4.4 cm
	Inner radius	4.0 cm
	Clad	Zr: 97.45%, Nb: 2.5%
Moderator/coolant	-	D_2O

[1]D. Albright, P. Brannan, and R. Kelley, "Mysteries Deepen Over Status of Arak Reactor Project," Institute for Science and International Security, (2009), <http://www.isisnuclearіran.org/assets/pdf/ArakFuelElement.pdf> (accessed 1 June 2011).

[2]D. Albright, P. Brannan, and R. Kelley, "Update on the Arak Reactor in Iran," Institute for Science and International Security, (2009), <http://isis-online.org/uploads/isis-reports/documents/Arak_Update_25_August2009.pdf> (accessed 1 June 2011).

[3]B. D. Murphy, "ORIGEN-ARP Cross-Section Libraries for the RBMK-1000 System," Oak Ridge National Laboratory, ORNL/TM-2006/139, (2006).

[4]U.S. Nuclear Regulatory Commission, "Report on the Accident at the Chernobyl Nuclear Power Station," NUREG-1250, (January 1987).

which have large neutron absorption cross sections. Ultimately, the reactor reaches an equilibrium state where production and removal of fission products are equal. At about 7,400 MWd/tHM, the reactivity drops below zero. The maximum burn-up for fuel assemblies can be calculated by using the method described in Driscoll et al.[34] This method is based on a polynomial approximation of the reactivity curve instead of a linear approximation normally used for light water reactors. Because of the small core size, a batch size of 10 is used for the subsequent calculations which corresponds to $B_d = 13.7$ MWd/kgHM. This estimation resembles that of Binford,[35] which mentions $n = 7...9$ as typical value for research reactors. The value of the maximum burn-up is relatively high for a HWR which is due to the simplifying reactivity model. In reality, it would likely be smaller for safety reasons and need to sustain a positive excess reactivity. But without knowing these details, the theoretical maximum is used for further calculations here.

The most relevant information for a possible proliferator is the total amount of plutonium produced with the given reactor and fuel cycle length

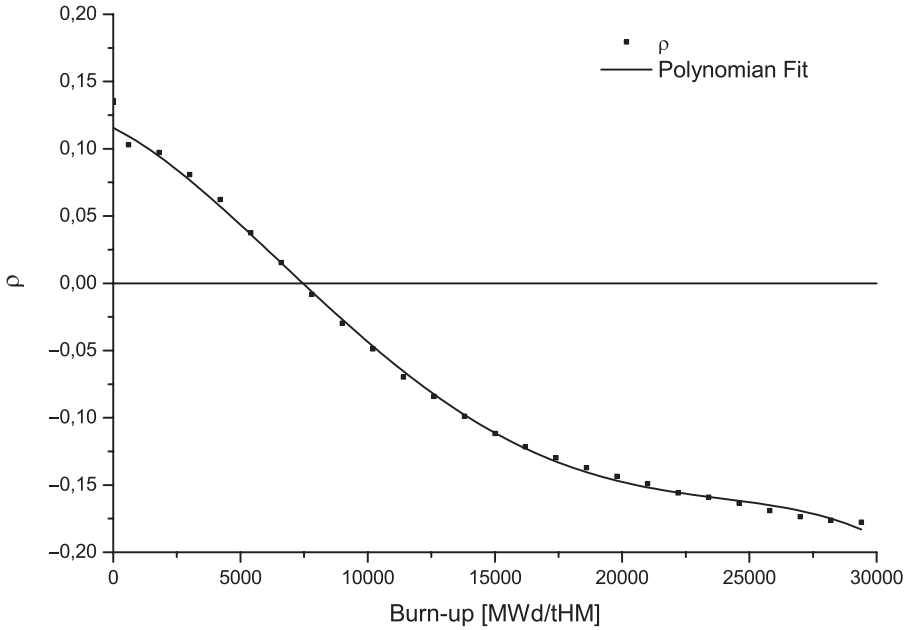

Figure 4: Reactivity as function of burn-up for the modeled fuel cell of Arak HWR. The coefficients for the polynomial fit are:
$\rho_0 = 0.116$, $A_1 = -9.60 \cdot 10^{-6}$, $A_2 = -1.38 \cdot 10^{-9}$, $A_3 = -8.92 \cdot 10^{-14}$, $A_4 = -1.46 \cdot 10^{-18}$.

as well as its fissile fraction. The average over the whole fuel burn-up with $B_d = 13.7$ MWd/kgHM is $C_B = 0.36$ g/MWd. Simultaneously, this is also the production averaged over the whole core because it consists of many batches of different burn-ups. In the first year of reactor operation, the production rate would certainly be higher because the whole core only consists of fresh NU rather than many batches of different burn-ups. Assume a specific power of $P_s = 20$ kW/kg, the burn-up which is achievable in one year is $B_a = 6.6$ MWd/kgHM for a load factor of 90 percent. For these burn-up intervals, C, and subsequently the plutonium production can also be calculated. Table 8 contains the results for annual plutonium production. Under optimal circumstances in the first year of operation (high load factor and discharge of low burn-up fuel), an output of 6.44 kg plutonium could be achieved. The production rates of 0.49 g/MWd–0.57 g/MWd for the first year fits well to the rate of 0.54 g/MWd given by IPFM for a NU-fuelled CANDU with a discharge burn-up of 7 MWd/kg.[36] The value calculated for steady-state operations assumes a higher discharge burn-up and is therefore lower. If the reactor is operated normally over a longer period with periodic change of fuel assemblies, the more realistic production rate is 3.94 kg/a, which is the value for moderate load factor and C averaged over the whole fuel cycle.

Table 8: Average plutonium amount produced in fuel assemblies in Arak HWR per year

Load factor (%)	First year of operation (kg/a)	Steady-state operation (C = 0.36 g/MWd) (kg/a)
60	5.00 (C = 0.57 g/MWd)	3.15
75	5.80 (C = 0.53 g/MWd)	3.94
90	6.44 (C = 0.49 g/MWd)	4.73

Note: The first column shows the production in the first year of operation when the whole core is composed of fresh NU-fuel (a specific power of 20 kW/kg is assumed to derive the discharge burn-up reachable in one year, which itself influences the production rate C). The second column gives the annual production rate under steady-state conditions.

These results imply a significantly lower production capability compared to the ones found in the references quoted above, which estimate a maximum capability of 8 kg/a–12 kg/a. The differences can be explained in two ways: First, the operator could increase the production by a more frequent fuel exchange and thus exploit the higher production rate at very low burn-ups. This question will be investigated further in scenario C since scenario A as analyzed herein assumes a normal reactor operation without any incentives of such frequent fuel exchange. Second, the literature values quoted above are only rough estimations of upper bound production capabilities of a 40 MW heavy water research reactor in general. It is not known which assumptions were made and therefore have limited reliability. Furthermore, the Arak reactor design may vary from the ones assumed in the literature thus leading to other production rates. In conclusion, the discrepancies found between the calculations presented here and in the literature do not refute the results.

If LEU fuel is used instead of NU, the production rates decrease according to the fuel enrichment. Table 9 shows the plutonium production rates for three different enrichments (3.5, 6, and 19.75 percent). The highest production rate can be achieved with 3.5 percent enriched fuel. At a typical discharge burn-up of 50 percent, the production rate is 0.23 g/MWd, which is less than half of the rate possible by using NU fuel.

Table 9: Plutonium production rates for scenario A and LEU fuel

Enrichment (% uranium-235)	Production rate at 40% burn-up (g Pu/MWd)	Production rate at 50% burn-up (g Pu/MWd)	Production rate at 60% burn-up (g Pu/MWd)
3.5	0.35	0.23	0.22
6.0	0.18	0.17	0.16
19.75	$1.4 \cdot 10^{-3}$	$1.3 \cdot 10^{-3}$	$1.3 \cdot 10^{-3}$

Table 10: Annual plutonium production at the Arak HWR. Low burn-up, 250 MWd/kg; medium burn-up, 500 MWd/kg; maximum burn-up for weapons-grade, 1,330 MWd/kg

Load factor (%)	Low burn-up (C = 0.79 g/MWd) (kg/a)	Medium burn-up (C = 0.77 g/MWd) (kg/a)	Max. burn-up for weapons-grade (C = 0.72 g/MWd) (kg/a)
60	6.92 kg/a	6.75 kg/a	6.31 kg/a
75	8.65 kg/a	8.43 kg/a	7.88 kg/a
90	10.4 kg/a	10.1 kg/a	9.46 kg/a

Results for Production Scenario C

In order to enhance the precision for the calculation of plutonium production rates at low burn-ups with NU fuel, smaller burn-up intervals are chosen (90 MWd/tHM instead of 1,200 MWd/tHM used in the previous calculations). The maximum burn-up required to ensure weapons-grade material is about 1,330 MWd/kg. This value seems reasonable: Albright assumes a value of 1,000 MWd/kg for weapons-grade plutonium production with HWRs.[37] C is determined at low (0–250 MWd/kg, $Pu_{fiss} > 98$ percent) and middle (0–500 MWd/kg, $Pu_{fiss} > 97$ percent) burn-up intervals, as well as for the whole irradiation period. With a moderate load factor and C averaged over the maximum burn-up, 7.9 kg could be produced annually (Table 10). With high load factor and very low burn-up (corresponding to a plutonium vector with ca. 1.5 percent non-fissile isotopes), the reactor would be capable of producing 10.4 kg/a. These values are relatively consistent with the lower range of the estimation on the Arak's production capability found in literature (8 kg/a–12 kg/a). Likewise, the production rate of 0.72–0.79 g/MWd agrees with IPFM for a HWR (0.78 g/MWd).[38]

Production Capabilities in States Not Part of the CTBT Annex II

In the context of nuclear non-proliferation, it is worthwhile to note the country's status within the Comprehensive Nuclear-Test-Ban Treaty (CTBT). This treaty forbids nuclear testing, provides for a global verification system, and is considered to be a major milestone in today's nuclear non-proliferation regime.[39] It was just entered into force since the provision that requires the 44 states listed in Annex II of the treaty to ratify it had not been fulfilled. These so called Annex II states are defined as members of the Conference on Disarmament in 1996 who formally participated in the CTBT negotiations *and* in 1996 had either power or research reactors. The intention of this provision was to guarantee universality of the treaty among the group of states that theoretically possess fissile material production capabilities *before* the treaty actually became legally binding. But according to the analysis presented here,

Table 11: Plutonium production in civilian research reactors in NNWSs and de-facto NWSs with a thermal energy below 1 MW

Country	Facility name	Plutonium production in fuel (g/a)	Plutonium production in NU targets (g/a)
Algeria	ES-SALAM	944	2,960
Argentina	RA-3	261	359
Australia	OPAL	522	717
Bangladesh	TRIGA MARK II	78.0	108
Belgium	BR-2	141	5,250
Brazil	IEA-R1	130	179
Bulgaria	IRT-SOFIA*	53	72
Canada	MNR MCMASTER, NRU	126	26,708
Chile	RECH-1, RECH-2*	182	251
Czech Republic	LVR-15 REZ	165	368
Egypt	ETRR-1, ETRR-2	626	860
Germany	BER-II	261	359
Greece	GRR-1*	130	179
Hungary	BUDAPEST RR	261	359
India	CIRUS, DHRUVA	27,580	27,580
Indonesia	TRIGA BANDUNG, GA SIWABESSY	834	1,152
Iran, Islamic Republic of	TRR, IR-40+	4,070	8,059
Israel	IRR-1	7	262
Japan	JRR-3M, KUR*, JRR-4, JMTR	2,043	2811
Jordan	JRTR+	130	179
Kazakhstan	WWR-K ALMA ATA, EWG 1	148	2,061
Korea, Dem. P. R. of	IRT-DPRK	209	287
Korea, Republic of	HANARO	11	5,930
Libyan Arab Jamahiriya	IRT-1	261	359
Morocco	MA-R1	52	72
Netherlands	HOR HFR	1,222	1,682
Norway	HBWR, JEEP II	1,057	4,330
Pakistan	PARR-1	261	359
Peru	RP-10	261	359
Poland	MARIA	782	1,080
Romania	PITESTI	365	502
South Africa	SAFARI-1	522	717
Taiwan	THOR	52	72
Thailand	TRR-1/M1	52	72
Ukraine	WWR-M KIEV	261	359
Uzbekistan	TASHKENT	261	359

Note: A load factor of 75 percent is assumed and the plutonium production rate is chosen corresponding to a burn-up of 50 percent uranium-235. The third column shows the annual production in the reactor fuel in normal operation (scenario A), the fourth column provides the production capability with NU targets (scenario B). Germany's FRM II is not included because of its unconventional core design.

*Temporary shut down.

†Under construction.

not all of the states which run research reactors today are included in the Annex II list. These are: Czech Republic, Greece, Jordan, Kazakhstan, Libyan Arab Jamahiriya, Morocco, Taiwan, Thailand, and Uzbekistan. Morocco's reactor went critical in 2007 and the Jordan JRTR is not built yet, but the other seven states actually possessed research reactors in 1996. The reason for not being listed in the CTBT Annex II is purely political: They were not members of the Conference on Disarmament in 1996, but rather observers to the negotiations. Table 11 shows that none of the states which are not part of the Annex II list of the CTBT has plutonium production capabilities that could be a source of concern (all under 0.5 kg/a). The intention behind Annex II of the treaty is therefore not generally undermined. The only exception is Kazakhstan with the possibility to produce over 2 kg/a by two reactors. This is the only case in which the original intention of the Conference on Disarmament, to guarantee that all countries with significant nuclear material production capabilities must be part of the treaty before entry into force, is undermined. Nevertheless, it should be noted that all of these states except for Thailand have already ratified the treaty.[40]

SUMMARY

The results presented here provide a transparent database on global plutonium production in civilian research reactors. It can serve as a source of technical background information for the assessment of a country's nuclear capabilities.

Using the plutonium production rates calculated here, the worldwide plutonium production capabilities in civilian research reactors can be evaluated (with limitations of adaptability as previously noted). Table 11 provides this overview for all light water and heavy water moderated research reactors in NNWSs and de-facto NWSs. A moderate load factor of 75 percent (which translates to 274 full power days per reactor per year) is assumed for all calculations as well as the fuel enrichment, as specified in Table 3. According to these calculations, 40 LWRs produce about 10.7 kg/a in their normal operation mode (scenario A). The two largest single unit (HFR, Netherlands and JMTR, Japan) are in fact capable of producing more than 1 kg/a, but the median of the distribution is only 136 g/a and the third quartile is found at 261 g/a. This rate is doubled in the case of weapons material production in NU targets (22.1 kg/a, scenario B). Naturally, the latter is purely theoretical providing an idea of the worldwide total production capability, without much relevance for reality because not all considered states would begin weapon's plutonium production at the same time. In this case, reactors in Belgium, Indonesia, Kazakhstan, and Poland exceed the 1 kg/a-threshold, but still 50 percent of the reactors are only capable of producing less than 180 g/a. The reactor unit with the highest fuel

production rate is JMTR in Japan (1.3 kg/a). The unit with the highest production capability with targets is the BR-2 in Belgium (5.2 kg/a).

Heavy water moderated reactors are capable of producing much more plutonium. The total annual production in NNWSs in normal fuel (scenario A) is 33.6 kg, the median of the distribution is about 940 g/a. These reactors have the capability to produce significantly more weapons material. In scenario C, the median moves to 6.9 kg/a, the third quartile to 10.8 kg/a. Iran, whose reactor was the representative model, would be capable of producing nearly one SQ per year (7.9 kg/a). For these calculations, a load factor of 75 percent is assumed and CIRUS in India (recently shut down) and Arak in Iran (under construction) are included. The reactor with the highest production rate for scenario A is located in India (DHRUVA, 19.7 kg/a). The one with the highest production capabilities in scenario C is the Canadian NRU (26.6 kg/a).

The practice of the IAEA to concentrate its safeguards on research reactors above a thermal capacity of 25 MW seems cautious enough compared to the results calculated here: An MTR with the same capacity LEU driver-fuel (19.75 percent enriched) and NU targets with a load factor of 75 percent would need around 9 years to produce one SQ of weapons-grade material (scenario B). The situation changes dramatically for an HWR, which needs only around 1.6 years for one SQ, if the whole reactor is fuelled with NU (scenario C, load factor of 75 percent and maximum burn-up for weapons-grade material assumed). In this case, IAEA's criterion that a reactor must be able to produce one SQ within one year of operation to justify additional safeguards is still not violated.

NOTES AND REFERENCES

1. International Panel on Fissile Materials, *Global Fissile Material Report 2010*, Princeton (2010) and International Panel on Fissile Materials, *Global Fissile Material Report 2011*, Princeton (2011). Panel on Fissile Materials, Global Fissile Material Report 2011, Princeton (2011).

2. In this work, Nuclear Weapon States are defined according to the non-proliferation treaty (China, France, Russian Federation, UK, USA), De-facto Nuclear Weapon States as India, North Korea, and Pakistan and Non-Nuclear Weapon States as all remaining states. Israel has not officially declared its status, but is widely believed to possess nuclear weapons. Therefore Israel is categorized as a de-facto Nuclear Weapon State while recognizing that its status is uncertain.

3. International Atomic Energy Agency, "Research Reactor Database," (2011), <http://nucleus.iaea.org/RRDB/> (accessed 1 June 2011).

4. G. Zuccaro-Labellarte, and R. Fagerholm, "Safeguards at Research Reactors: Current Practices, Future Directions," *IAEA Bulletin* 4(1996): 20–24.

5. F. T. Binford, "Diversion Assumptions for High-Powered Research Reactors," Oak Ridge National Laboratory, ISP C-50 Phase I (1984).

6. G. Zuccaro-Labellarte, and R. Fagerholm, *op. cit*.

7. M. M. Miller, and C. A. Eberhard, "The Potential for Upgrading Safeguards Procedures at Research Reactors Fuelled with Highly Enriched Uranium," Massachusetts Institute of Technology (1982).

8. U.S. Department of Energy, "Plutonium: The First 50 Years. United States Plutonium Production, Acquisition, and Utilization from 1944 through 1994," DOE/DP-0137, (1996).

9. J. C. Mark, "Explosive Properties of Reactor Grade Plutonium," *Science and Global Security*, 4(1993): 111–128.

10. "50-Yr Old Research N-Reactor CIRUS Shut Down," Outlook India, 31 December 2010, news.outlookindia. com/items.aspx?artid=706959.

11. International Panel on Fissile Materials (2010), *op. cit.*

12. Oak Ridge National Laboratory, *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation*, ORNL/TM-2005/39 (2009).

13. D. Bodansky, *Nuclear Energy: Principles, Practices and Prospects* (New York: Springer, 2004).

14. J. R. Larmash, and A. J. Baratta, *Introduction to Nuclear Engineering* (Prentice Hall, 2001).

15. Assumptions for CANDU: pitch (square) of 28.6 cm; radius of fuel rod: 0.605 cm; 37 rods per assembly. Assumptions for NRU: pitch (hexagonal) of 19.7 cm; radius of fuel rod: 0.548 cm, one rod per assembly.

16. International Atomic Energy Agency, *Research Reactor Core Conversion from the use of Highly Enriched Uranium to the use of Low Enriched Uranium Fuels Guidebook*, TECDOC-233, (Vienna: IAEA, 1980).

17. International Atomic Energy Agency (1980), *op. cit.*

18. F. T. Binford, *op. cit.*

19. M. M. Miller, and C. A. Eberhard, *op. cit.*

20. *Ibid*.

21. F. T. Binford, *op. cit.*

22. For example, the production rate at 60 percent burn-up for LEU fuel is 89 mg Pu/MWd compared to 100 mg Pu/MWd given by A. Glaser, "On the Proliferation Potential of Uranium Fuel for Research Reactors at Various Enrichment Levels," *Science and Global Security*, 14(2006): 1–24. However, a detailed comparison is not possible due to the fact that the calculations by Glaser assume a different uranium density in the fuel (0.948 gU/cm3 for all enrichment levels).

23. F. T. Binford, *op. cit.*

24. International Atomic Energy Agency (1980), *op. cit.*

25. M. M. Miller, and C. A. Eberhard, *op. cit.*

26. *Ibid.*

27. A. Tomanin, P. Peerani, and G. Janssens-Maenhout, "Pu-Breeding Feasibility in Irradiation Channels of Research Reactors," in: *Proceedings of the IAEA Safeguards Symposium 2010*, International Atomic Energy Agency, IAEA-CN-184/15 (2010),

28. See for example: International Atomic Energy Agency, "Implementation of the NPT Safeguards Agreement and Relevant Provisions of Security Council Resolutions in the Islamic Republic of Iran," GOV/2010/62, (23 November 2010).

29. GlobalSecurity, "Arak Iran Nuclear Research Reactor (IR-40)," GlobalSecurity.org, (2011), <http://www.globalsecurity.org/wmd/world/iran/arak-nrr.htm> (accessed 1 June 2011).

30. R. Prasad and J. M. Parillo, "Iran's Programs to Produce Plutonium and Enriched Uranium," Carnegie Endowment for International Peace, (2006), <http://www.carnegieendowment.org/static/npp/Iran_fact_sheet.pdf> (accessed 1 June 2011).

31. D. Albright, P. Brannan, and R. Kelley, "Mysteries Deepen Over Status of Arak Reactor Project," Institute for Science and International Security, (2009), <http://www.isisnuclearіran.org/assets/pdf/ArakFuelElement.pdf> (accessed 1 June 2011).

32. D. Albright, P. Brannan, and R. Kelley, "*Update on the Arak Reactor in Iran*," Institute for Science and International Security, (2009), <http://isis-online.org/uploads/isis-reports/documents/Arak_Update_25_August2009.pdf> (accessed 1 June 2011).

33. M. J. Driscoll, T. J. Downar, and E. E. Pilat, *The Linear Reactivity Model for Nuclear Fuel Management* (La Grange Park: American Nuclear Society, 1990).

34. If an operator wants to utilize fuel as economically as possible, he will have to consider in-core fuel management strategies like a batch reloading scheme in order to keep the "reactivity swing" (positive excess reactivity of the core) as low as possible. Thus, a batch reloading strategy will be included in the calculation of discharge burn-up as described in the following (based on Driscoll).

If a reactor core consists of n batches, 1/n of the core will be removed after one fuel cycle. If the reactor has online refueling capability, as Arak probably has, this strategy will be exercised to a large extend. For LWRs the reactivity ρ is in good approximation a linear function of burn-up. For calculations of reloading patterns the so called linear reactivity model is then applied:

$$\rho(B) = \rho_0 - A \cdot B,$$

where B is the burn-up and ρ_0 the extrapolated initial reactivity of the fresh fuel after saturation of fission products has come to equilibrium. In the case of heavy water reactors like Arak, the reactivity shows no such linear trend. A nonlinear approximation has to be found in order to derive the cycle burn-up B_c as a function of the number of batches n. Assume a nonlinear reactivity curve $\rho(B)$, which can be approximated by a polynomial of this form:

$$\rho(B) = \rho_0 + A_1 B + A_2 B^2 + A_3 B^3 + A_4 B^4.$$

The reactor core consists of n fuel batches. With every refueling ("cycle"), the batch with the highest burn-up is removed and a new batch with fresh fuel is inserted. If the burn-up of one cycle is Bc, the discharge burn-up then is

$$B_d = n \cdot B_c.$$

The reactivity for batch j, that was in the core for j cycles, is given by

$$\rho_j(B) = \rho_0 + A_1 B_j + A_2 (B_j)^2 + A_3 (B_j)^3 + A_4 (B_j)^4.$$

One can estimate the reactivity for the whole core by calculation of the sum over all batches present in the core and setting the resulting equation equal to zero:

$$\sum_{j=1}^{n} \rho_j(B) = n\rho_0 + A_1 B_c \sum_{j=1}^{n} j + A_2 B_c \sum_{j=1}^{n} j^2 + A_3 B_c \sum_{j=1}^{n} j^3 + A_4 B_c \sum_{j=1}^{n} j^4 = 0.$$

Summation formulas can be used for evaluating the sums over j. The equation then can be solved numerically for B_c. The theoretical limit value for B_d is reached with $n = \infty$, and is equal to the relation:

$$\int_0^{B_d} \rho(B)dB = 0.$$

This model allows an approximate estimation of B_d, but makes certain simplifications: It is assumed that reactivity is equal to the whole core and that reloading patterns are not influenced by economic or operating aspects regarding fuel cycle or research purposes. Since such information is not available, these simplifications have to be accepted. For real reactors, the number of batches is of course limited, the maximum would be the number of the smallest single unit, which can be exchanged (e.g., a fuel assembly). For Arak, this maximum would be $n = \frac{m_{fuelcell}}{m_{core}} = 122.4\frac{kgHM}{m_{core}}$ For $m_{core} = 5tHM$ n would be equal to 41. But other constraints like maintenance, reactivity limitations imposed by safety regulations, the experimental program, and economic considerations will reduce n further.

35. F. T. Binford, *op. cit.*

36. International Panel on Fissile Materials (2010), *op. cit.*

37. D. Albright, F. Berkhout, and W. Walker, *Plutonium and Highly Enriched Uranium 1996: World Inventories, Capabilities and Policies*, (Oxford University Press Inc., 1997)

38. International Panel on Fissile Materials (2010), *op. cit.*

39. Comprehensive Nuclear-Test-Ban Treaty, 1996.

40. Preparatory Commission for the Comprehensive Nuclear-Test-Ban Treaty Organization, *Status of Signature and Ratification*, <http://www.ctbto.org/the-treaty/status-of-signature-and-ratification/?Fsize=a> (accessed 1 June 2011).

41. International Atomic Energy Agency (2011), *op. cit.*