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Feasibility and Desirability of Employing the Thorium Fuel Cycle for Nuclear Power Generation

Bal Raj Sehgal
Professor Emeritus, Nuclear Power Safety (NPS)
Royal Institute of Technology (KTH)
AlbaNova
SE-106 91 Stockholm
e-mail: bsehgal@safety.sci.kth.se

Introduction & Background - 1

- Thorium was discovered by the Swedish chemist Jakob Berzelius in 1828. The only commercial use of Thorium has been in gas mantles. Natural Thorium does not contain a fissile element as does Uranium (0.7% U-235). Thorium on capture of a neutron becomes U-233, which is a very efficient fissile fuel (η=2.06) for thermal reactors.
- In late 1950s and early 1960s, breeder development was being pursued in USA through the fast reactor employing Pu and U. The Pu was to be generated from LWRs, however the future was to be LMFBRs, which could sustain nuclear energy for hundreds of years.
- Dr. Radkowsky, the chief scientist of US Nuclear Navy came up with the idea of developing a breeder reactor with water technology (since the U.S. navy was developing it for submarines). He developed a high neutron economy core design called LWBR, currently called the Radkowsky Reactor Design which employed Th-232 and U-233.
- The LWBR was constructed and operated for a few cycles. The core was reprocessed and a breeding ratio of ~1.01 was obtained, thereby fissile breeding in the thermal reactor thorium cycle was confirmed.

Introduction & Background - 2

- The PWR and BWR with U cycle were already developed. The electric power industry did not want to wait for the development of LWBR. The Uranium was cheap ($8.00/lb or $18/Kg) and more U was found. Thus Th fuel cycle development work was abandoned. However, work continued at a low level (with my participation). In particular, critical experiments were conducted.
- The Thorium work was also picked up in other countries, e.g. India, since it had large deposits of Thorium, but not of Uranium.
- In early 70’s with plant orders reaching values of more than a hundred in USA, it appeared that Uranium could get very expensive. This led to a revival of some Thorium work (in which I also participated). An assessment of Thorium use in PWRs was published.
- The orders of LWRs stopped world-wide after the TMI-2 and Chernobyl accidents; and the pressure on Uranium supply decreased very much. Thorium development never got encouraged, even with the curtailment of the fast reactor programs world-wide.
- Thorium-fuelled FBRs were investigated, with ideas about cross-progeny fuel cycles, in which U-233 was bred in LMFBRs and employed in water or heavy water reactors to obtain high efficiency. However, no Thorium fuelled LMFBR was built.

Thorium Fuelled Reactors-1

- In spite of the fact that U fuelled power reactors were dominant; several small-scale reactors were constructed with Thorium fuel. The higher power commercial reactors were the PWR at Indian Point New York, the HTGR at Fort St. Vrain and the Pebble Bed Gas Reactor THTR in Germany.

Thorium Fuelled Reactors-2

Table 1. Reactors operated with substantial amounts of thorium

Reactor	Type	Location	Operating period(with thorium)	Thermal power (MWt)	Electric power (MWe)
Elk River	BWR	USA	1962-1968[9]	58	15*
Indian Point	PWR	USA	1962-1964[9]	615	151
Peach Bottom 1	HTGR-prismatic	USA	1966-1972[8]	115	40
Shippingport	LWBR**	USA	1977-1982[8]	236	60
Fort St Vrain	HTGR-prismatic	USA	1974-1989[9]	842	330
AVR	HTGR-pebble	Germany	1967-1988[8]	45	15
THTR	HTGR-pebble	Germany	1985-1989[8]	750	300
Various	PHWR	India	Ongoing	–	–
Dragon	AGR	England	1966-1973	20	
FBTR	LMFBR	India	Ongoing	40	

* plus additional from fossil-fired super heaters.
** Data represent LWBR core, not original Shippingport core, which did not use thorium or ^{233}U

Thorium Fuelled Reactors-3

- The Elk River Plant operated with 93% U-235 and Th fuel in S.S. Clad. The plant shut down after 6 years due to stress-corrosion cracking. No problem with fuel. There was no reprocessing of fuel.
- The Indian Point-I Reactor used the same fuel as Elk-River Plant, for its first core. The reactor operation was normal. The reactor changed to the regular LWR fuel (3% enriched Uranium) after first core due to cost of the highly enriched Uranium.
- The final core of the Shipping Port Reactor employed U-233-Th LWBR core. The LWBR core employed seed-blanket assemblies with seed containing U-233-Th and blanket containing Th fuel. The seed was moveable in order to provide reactivity control; and the blanket part was stationary. The cladding was Zircaloy and no poison control rods were employed. Thus it was a very highly neutron-economic reactor. No safety or licensing issues were noted. The irradiated fuel from this plant was reprocessed at the NFS plant in New York. The fuel burn-up was not very high. The reprocessing established that the LWBR core did breed (BR~1.01).

Thorium Fuelled Reactors-4

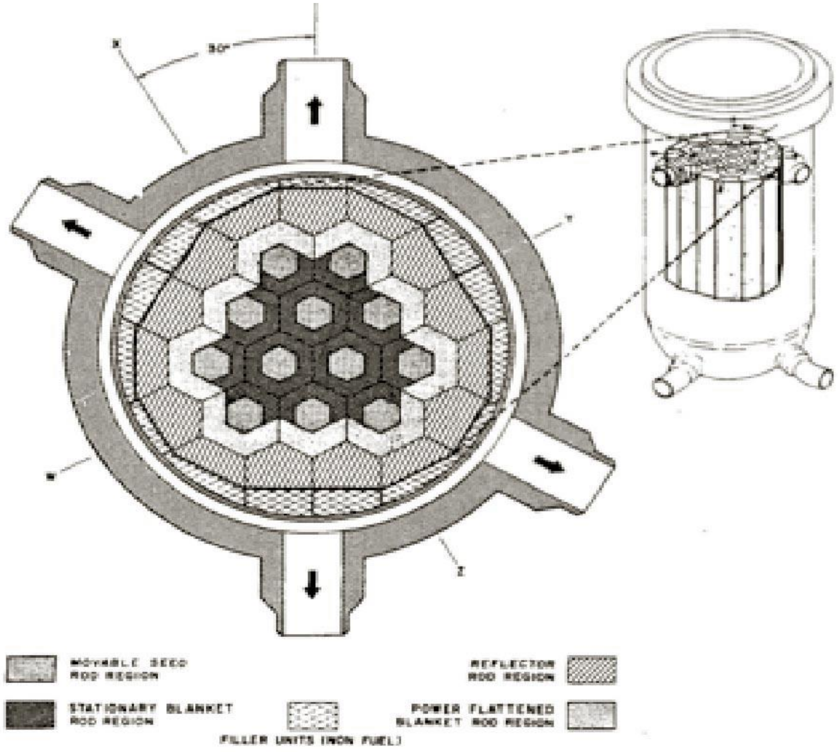


Figure 1. Shippingport LWBR Core with Hexagonal Seed-Blanket Assemblies

Thorium Fuelled Reactors-5

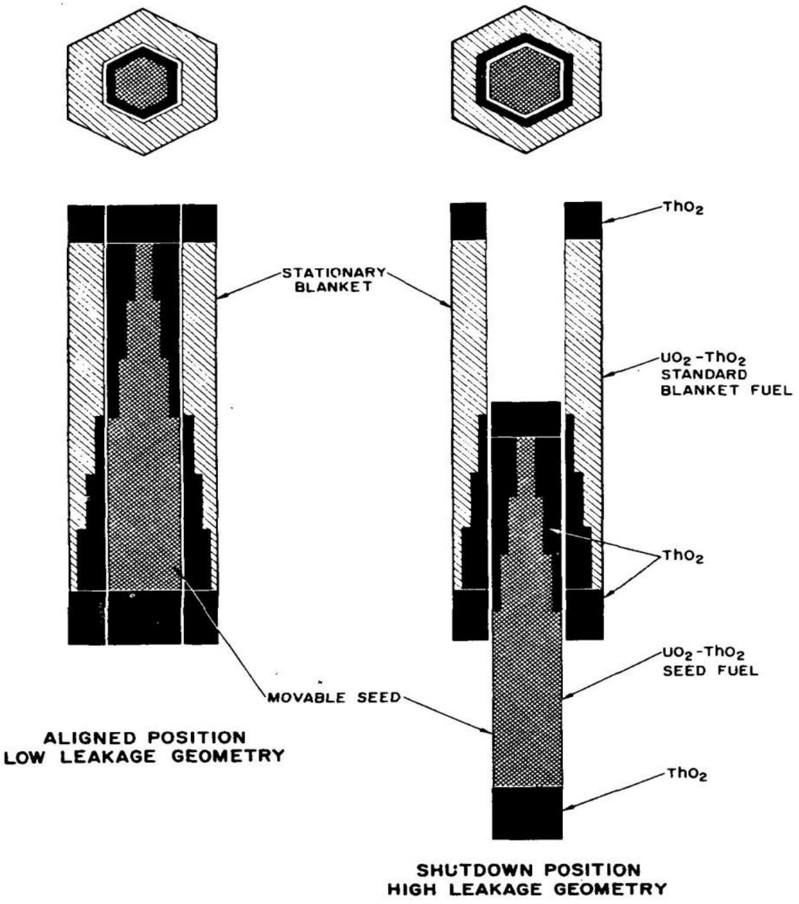


Figure 2. Control of LWBR by Vertical Movement of Seed Module

Thorium Fuelled Reactors-6

- India has irradiated Thorium rods and bundles in CANDU type heavy water reactors. There is no indication of any operational or safety issues. Perhaps, the fuel has been processed.
- Thorium 93% enriched Uranium was the fuel in Peach Bottom and Fort St. Vrain prismatic core gas-cooled reactors; constructed in USA and the THTR, pebble bed gas-cooled reactor in Germany. The Triso fuel worked well for even very high burn-up. The Fort St. Vrain reactor operated with Carbide fuel kernels for 15 years. The plant was shut down due to problems with Helium circulators. No problem was found with fuel. The THTR pebble beds released some fission gases and some pebbles were found to have incurred damage at high burn-up.

Thorium Fuelled Reactors-7

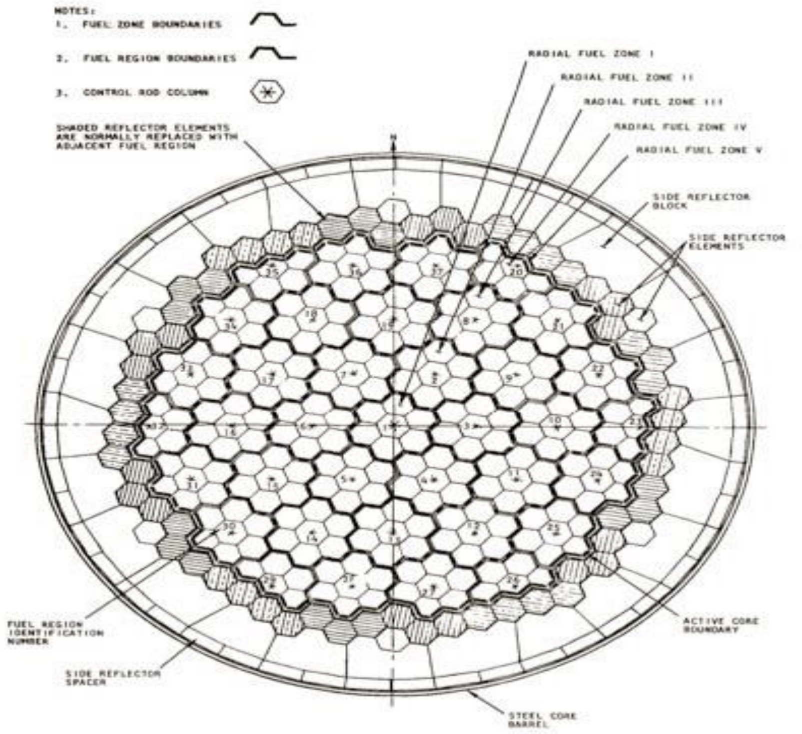


Figure 3. Fort St Vrain Core Layout

Thorium Fuelled Reactors-8

Table 2. Operating Experience with Thorium/ ^{233}U

Reactor	Operating life with thorium	Fuel reprocessed for recycle?*	Operated mostly with ^{233}U?	Fresh fuel with ^{233}U handled at reactor
Elk River	7 yrs	No	No	No
Indian Point 1	3	No	No	No
Peach Bottom 1	7	No	No	No
Shippingport	6	No	Yes	Yes
AVR	15	No	No	No
THTR	22	No	No	No
PHWR	Ongoing	Pending	No	No
Dragon	8	No	No	No

* Reprocessed as a matter of course, with recycle of recovered ^{233}U

Thorium Fuelled Reactors-9

Summarizing, the experience base of Thorium fuel in thermal and fast reactor is relatively meager compared to that of U fuel cycle. In addition in all the plants, except for the LWBR plant, the core did not reach equilibrium fuel conditions, so that the low value of beta with U-233 did not create any safety issues. This kind of issue is also faced by core containing Pu fuel. The LWBR operators did not report any problem with low value of beta. All the Th fuelled reactors employed open cycle, except that the LWBR employed recycled U-233 fuel, obtained from other reactors.

Back End of Thorium Fuel Cycle -1

- It has been clear from the very beginning of the Thorium utilization history that the full benefit of employing Thorium fuel is attained only after the back end (reprocessing and refabrication) is established and U-233 is used.
- LWBR and Indian Point fuel was reprocessed at the NFS plant in New York State. In addition considerable development work was performed in USA on reprocessing and refabrication of Th-U233 fuel. Pilot-to-large-scale facilities were constructed and operated. Most of the pilot scale work was performed in ORNL and B&W. However, reprocessing work was also performed in Hanford and Savannah River U-Pu reprocessing plants. The development work was very active in 1960s, but it was stopped in 1970s.
- The Acid-Thorex process with solvent extraction, which is similar to the Purex process for extraction of Pu, was employed at all facilities for extraction of U-233. The head-end process, developed at ORNL, was claimed to be ready for industrial application. The cost was estimated to be 30% larger than that for U reprocessing. The NFS cost was published as $78/Kg of reprocessed Th fuel.
- A close-coupled pilot plant for reprocessing and refabrication was constructed in ORNL in a shielded facility. The reprocessed product is a sol-gel which can be directly employed for refabrication. This facility called Kilorod Facility produced 1000 rods for criticality experiments at BNL. These rods were handled quite easily.

Back End of Thorium Fuel Cycle -2

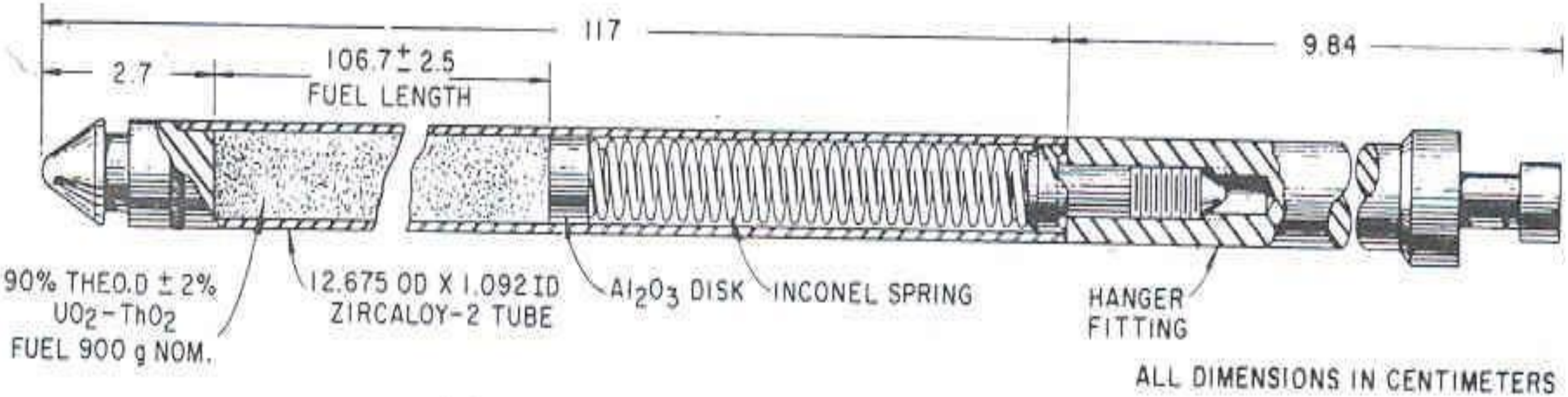


Figure 5. Design features of the BNL fuel rod

Back End of Thorium Fuel Cycle -3

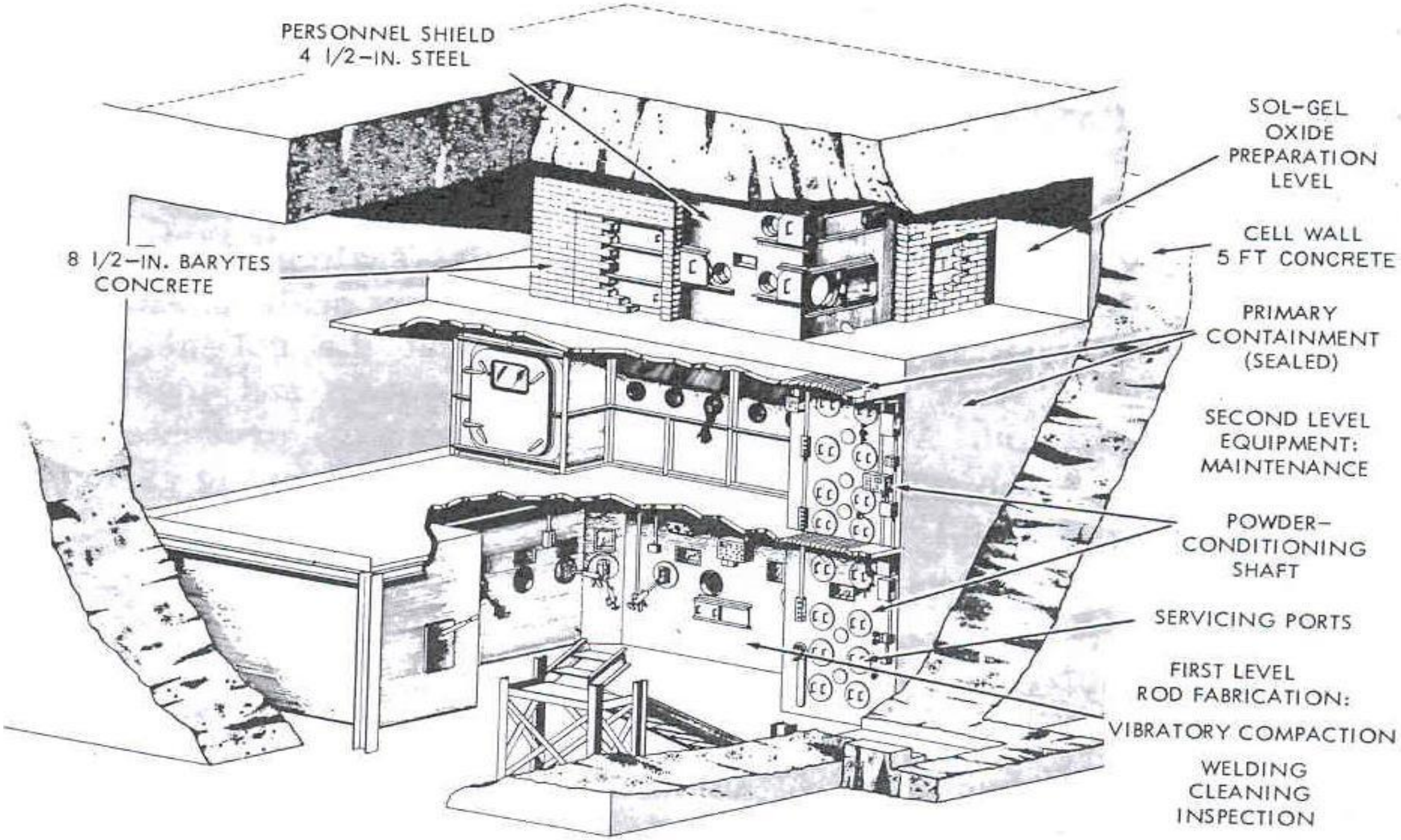


Figure 6. Kilorod Facility

Back End of Thorium Fuel Cycle -4

- The major problem in the refabrication part of the back-end of the Th fuel cycle is the radiation dose from U-232, which builds up as time progresses. If the content of U-232 in U-233 is less than 1000 PPM, rapid fabrication could be performed in unshielded facilities. B&W demonstrated the construction of one fuel assembly with 3% U-233 and 97% ThO_2. The U-233 they employed was freshly separated and had ~42 PPM U-232 and all personnel doses were within the acceptable limits. The sol-gel derived powder was added to clad material cylinders which were vibratory-compacted while residing in a water pool and later welded on top.
- It was found that the radiation dose hzard is much less if Th-228 is seperated immediately from U-233. An interesting set of pilot scale tests were performed at Hanford and Savannah River Pu-reprocessing plants to produce U-233 having less than 5 PPM U-232. They produced 160 Kg of 233. This product could be handled (for fabrication) with glove-box operation as practiced for the U-PuO_2 fuel fabrication.
- A coupled reprocessing-refabrication facility called PCUT was constructed in Italy. This was a remotely-operated shielded facility, since it was argued that the flexibility to wait between reprocessing and refabrication would be needed. Unfortunately, the remote operation had troubles and the refabrication part was shut-down. PCUT reprocessed the irradiated fuel from the Elk-River plant in 1975.

Back End of Thorium Fuel Cycle -5

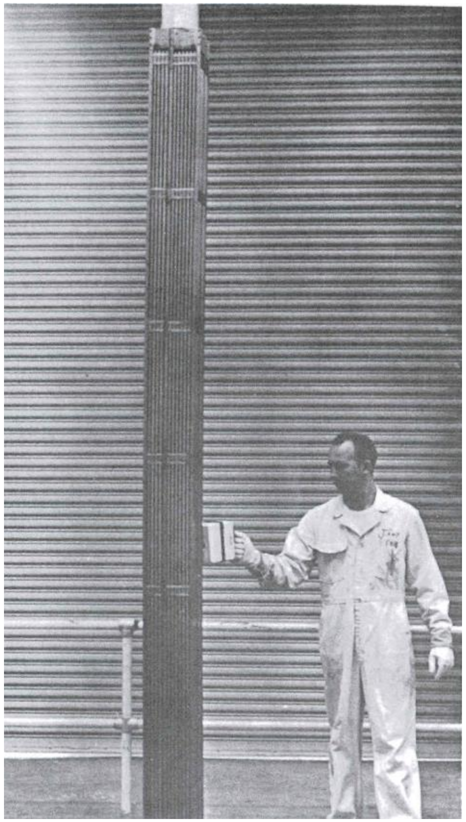


Figure 4. Finished fuel assembly produced during demonstration program (withdrawn rapidly from pool for quick survey of gamma activity)

Back End of Thorium Fuel Cycle -6

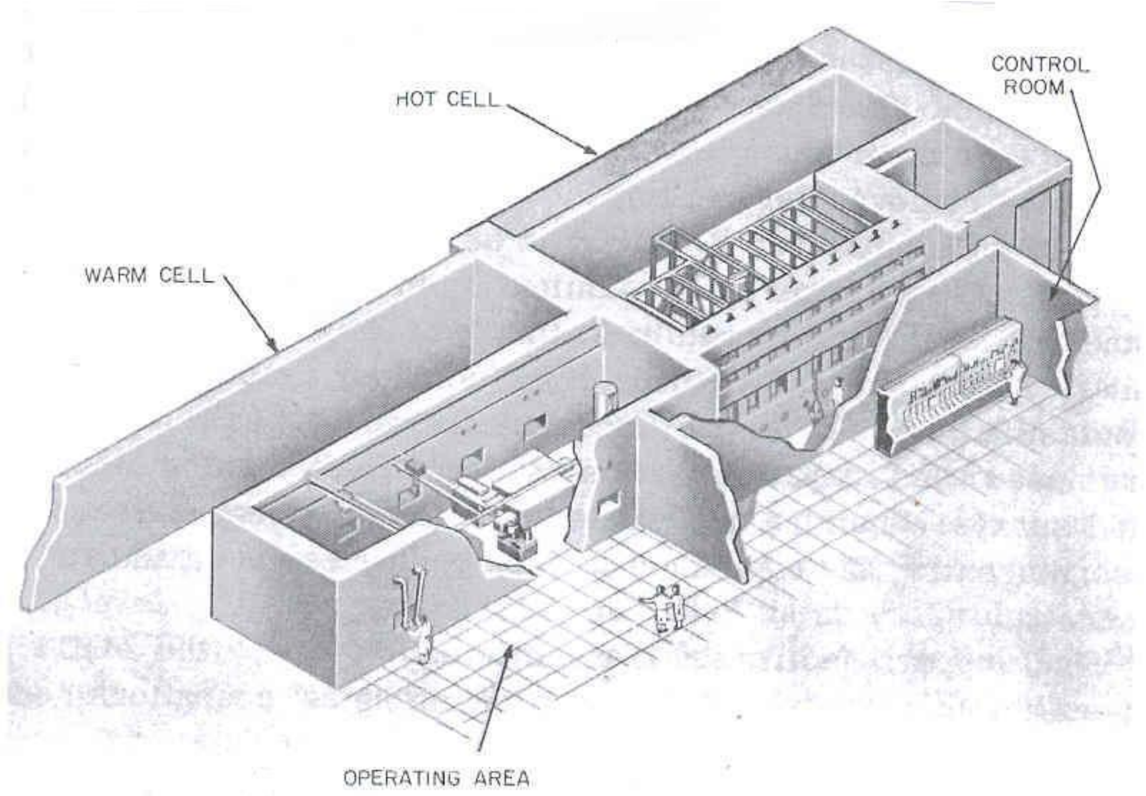


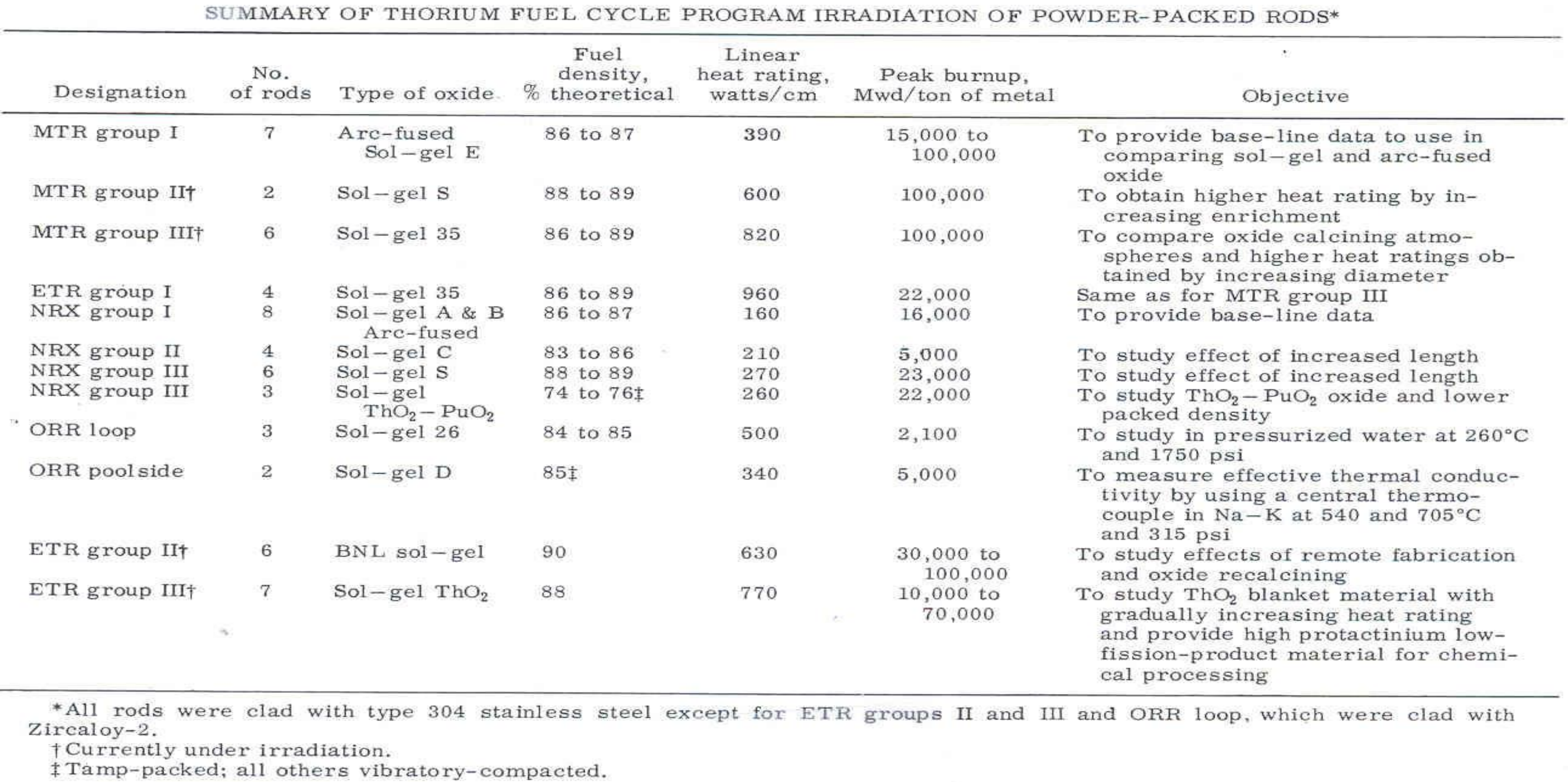
Figure 7. Shielded areas in the PCUT facility

Irradiation Experience of Thorium-Based Fuel -1

- It is imperative that the Th-based fuel perform at least as well as the U-based fuel in the reactor. ThO_2 has been found to be a more stable compound than UO_2, however, it is important to establish the burn-up level that it can sustain, and the duty cycle the fuel can endure.
- ORNL conducted a long campaign of irradiating samples of sol gel-derived pure and mixed components of Th, U and Pu as oxides and carbides. PNL and Uranium institute irradiated Zircaloy clad ThO_2 - PuO_2. They performed stringent (100 KW/m) duty cycle and long burn-up (~110,000 MWd/tonne) tests. Some of the samples experienced center melting.
- The experience with these irradiations was positive in that the behavior of the Th-based fuels was as good or better than that of U-based fuels.
- Reactor physics analyses of Th-fuelled cores has indicated that with high burn-up, the U-233 formed adds reactivity and a considerable fraction of U-233 formed can be burned in situ. The once-through (open) Thorium fuel cycle could, in fact, compete with the once-through Uranium fuel cycle in economics, in spite of needing greater enrichment at the beginning of the fuel cycle.

Irradiation Experience of Thorium-Based Fuel -2

Table 3



SUMMARY OF THORIUM FUEL CYCLE PROGRAM IRRADIATION OF POWDER-PACKED RODS*

Designation	No. of rods	Type of oxide	Fuel density, % theoretical	Linear heat rating, watts/cm	Peak burnup, Mwd/ton of metal	Objective
MTR group I	7	Arc-fused Sol-gel E	86 to 87	390	15,000 to 100,000	To provide base-line data to use in comparing sol-gel and arc-fused oxide
MTR group II†	2	Sol-gel S	88 to 89	600	100,000	To obtain higher heat rating by increasing enrichment
MTR group III†	6	Sol-gel 35	86 to 89	820	100,000	To compare oxide calcining atmospheres and higher heat ratings obtained by increasing diameter
ETR group I	4	Sol-gel 35	86 to 89	960	22,000	Same as for MTR group III
NRX group I	8	Sol-gel A & B Arc-fused	86 to 87	160	16,000	To provide base-line data
NRX group II	4	Sol-gel C	83 to 86	210	5,000	To study effect of increased length
NRX group III	6	Sol-gel S	88 to 89	270	23,000	To study effect of increased length
NRX group III	3	Sol-gel ThO_2-PuO_2	74 to 76‡	260	22,000	To study ThO_2-PuO_2 oxide and lower packed density
ORR loop	3	Sol-gel 26	84 to 85	500	2,100	To study in pressurized water at 260°C and 1750 psi
ORR poolside	2	Sol-gel D	85‡	340	5,000	To measure effective thermal conductivity by using a central thermocouple in Na-K at 540 and 705°C and 315 psi
ETR group II†	6	BNL sol-gel	90	630	30,000 to 100,000	To study effects of remote fabrication and oxide recalcining
ETR group III†	7	Sol-gel ThO_2	88	770	10,000 to 70,000	To study ThO_2 blanket material with gradually increasing heat rating and provide high protactinium low-fission-product material for chemical processing

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*All rods were clad with type 304 stainless steel except for ETR groups II and III and ORR loop, which were clad with Zircaloy-2.

†Currently under irradiation.

‡Tamp-packed; all others vibratory-compacted.

Thorium Fuel Cycle in LWRs-1

- Thorium use together with 20w% enriched Uranium has been studied in some detail. The Uranium requirements are greater if the fuels are mixed together. It has been calculated that with homogenous mixture, Th-U fuel can compete with once-thru U cycle at burn-up levels of ~120 MWd/te in PWRs. For pressurized heavy water reactors the competing burnup levels would be 40 – 50 MWd/te. Both of these burnup are approximately twice the current limit for these plants. For 20% enrichment in U, 75% of fuel could be Thorium.

- Heterogeneous designs are based on the Radkowsky patented Thorium fuel design. The seed-blanket concept can be incorporated in a square PWR assembly called SBU, or separate, seed and blanket assemblies (WASB). The WASB raises less issues about mechanical loads and avoids moving of seed assemblies. The operational strategy for heterogeneous designs is to replace seeds at normal intervals of 4 to 6 years, while keeping the blanket regions for 8-12 years, allowing time to fission most of the U233 formed.

- Each of the seed blanket variants is subject to some licensing issues:

  - burn-up greater than currently achieved
  - seed power much greater than blanket power needing different fuel pin designs and water flows with time
  - reduced effectiveness of control rods and boron.

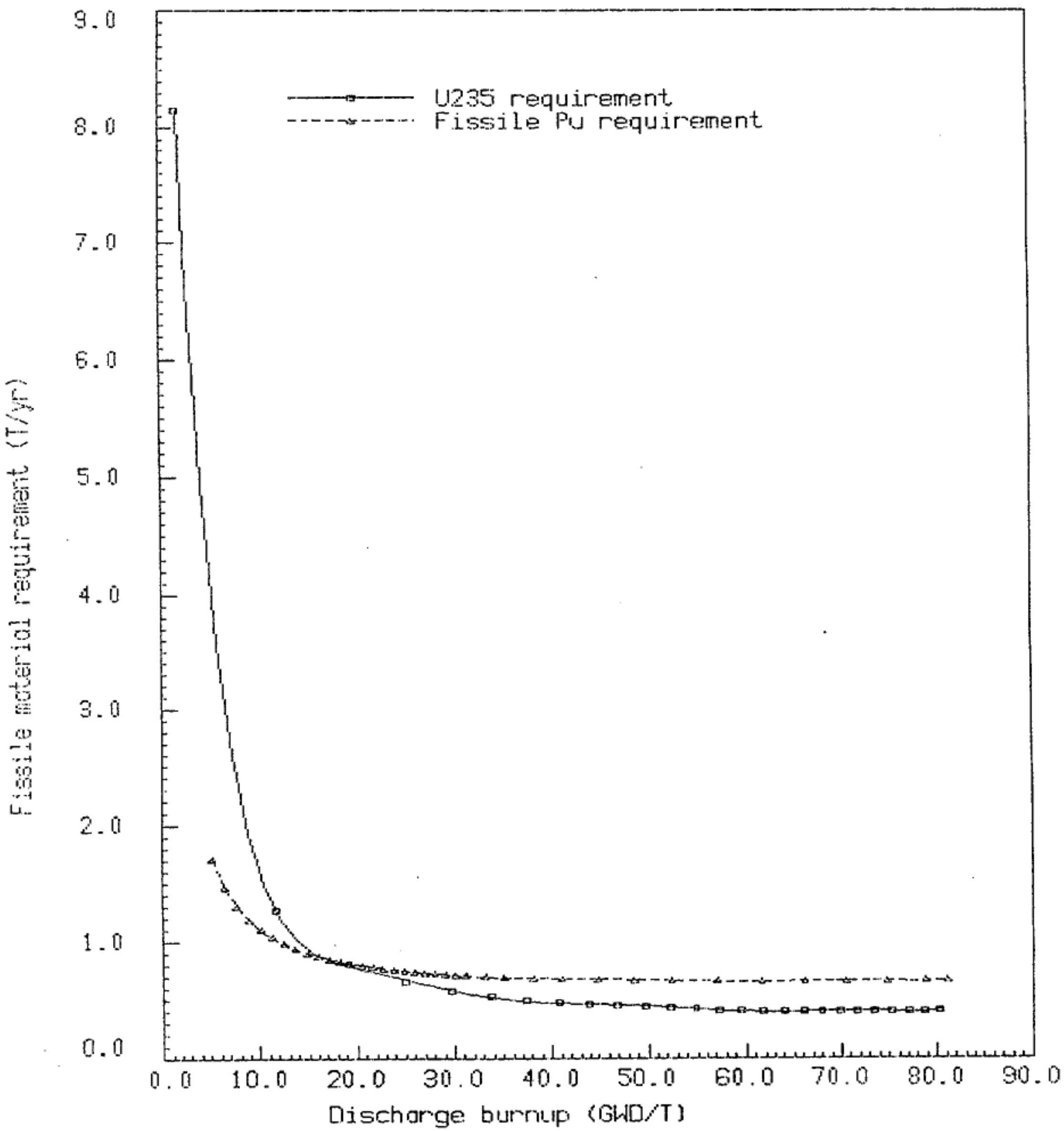


FIG. 3.4.2. Fissile requirement with once through cycle for a 750 MW(e) PHWR..

Thorium Fuel Cycle in LWRs-2

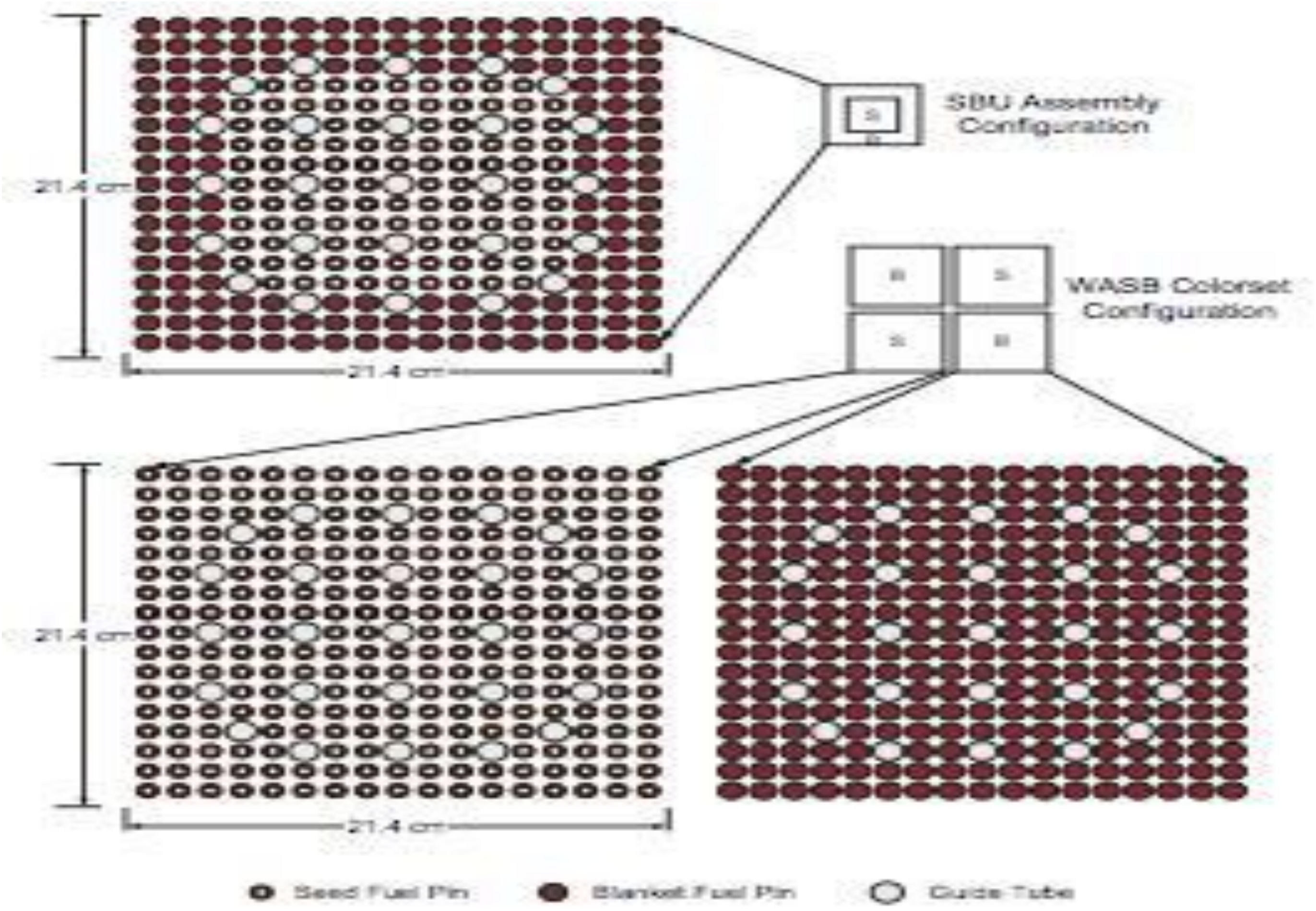
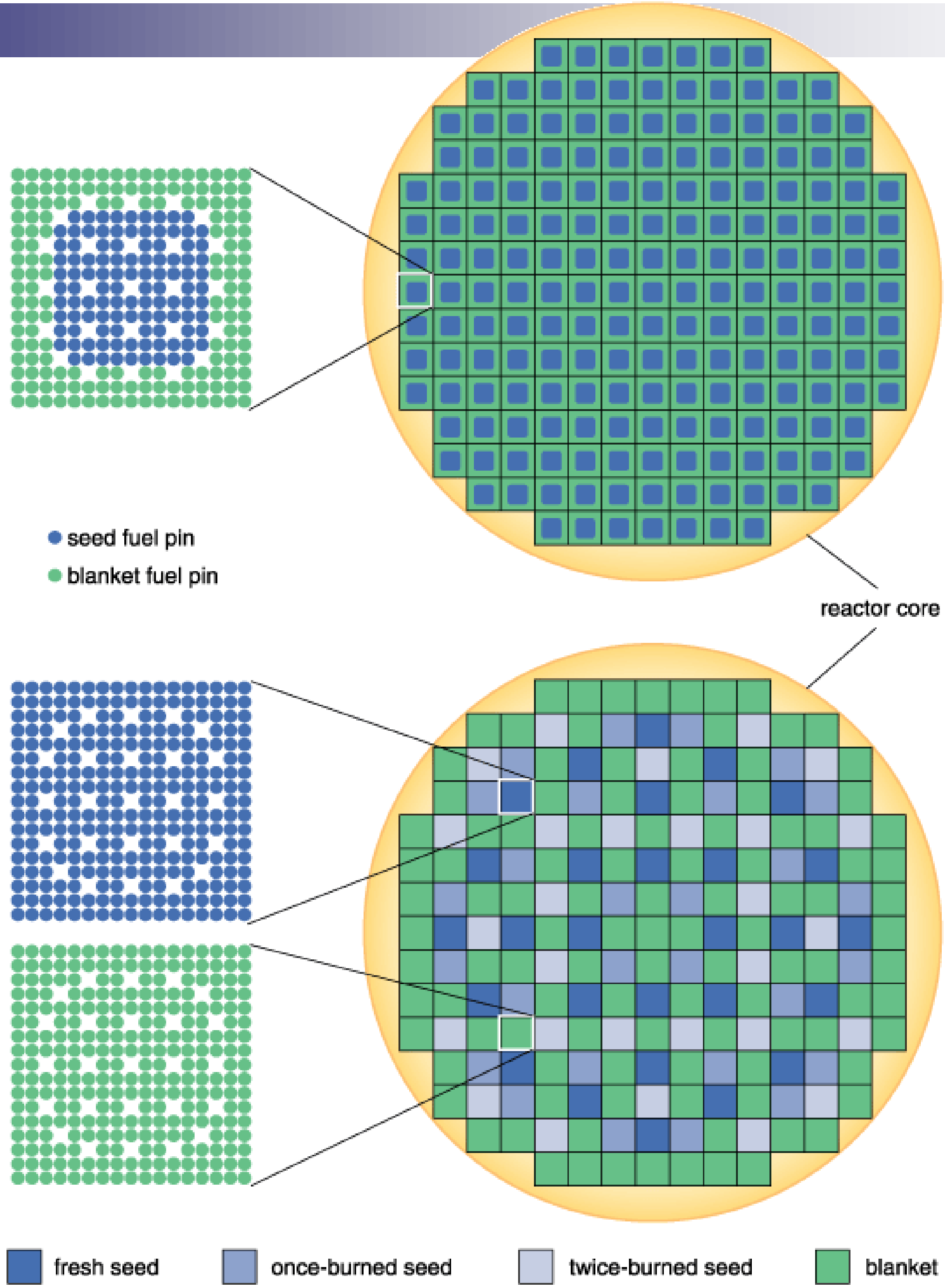


Figure 8. Layout of SBU and WASB Fuel Assembly Designs


● seed fuel pin
● blanket fuel pin
reactor core
fresh seed
once-burned seed
twice-burned seed
blanket

Desirability of Establishing the Thorium Fuel Cycle for Nuclear Power Generation -1

We can enumerate the advantages and disadvantages of Th-based fuels for nuclear power generation.

<u>Advantages</u>

- U-233 is the best thermal spectrum fuel.
- ThO_2 has better stability and conductivity than UO_2.
- Th will not produce as long-life actinides as U-238. Waste volumes will also be smaller.
- Th is excellent fuel for burning Pu in both thermal and fast reactors. In fast reactors it may provide negative void coeff.
- The radioactive U-232 in U-233 provides resistance to proliferation.
- The high decay heat of Pu produced if medium enriched U is used to start Th cycle, also provides resistance to proliferation.
- Much U-233 formed can be burned if high burn-ups are achieved, which may be feasible with ThO_2.
- ThO_2 should be cheaper than UO_2 on world market.
- Th is more abundant and wide-spread. Its price may not escalate as much as that of UO_2.

Desirability of Establishing the Thorium Fuel Cycle for Nuclear Power Generation -2

Disadvantages

- Natural Thorium has no fissile isotope. It has very low value of fast fission cross section.
- Th irradiation produces U-232 making the fabrication of fuel containing U-233 difficult ad expensive.
- Th enriched with MEU will produce high decay heat Pu, making storage and disposal more difficult.
- Thorium produces Protactinium which decays to U-233 with 27 day half life. This requires greater shut-down requirements for 3 months of outage.
- U-233 and Pu-239 provide a Beta of ~0.3% versus 0.5% for U-235; reactivity transients will be faster.
- Control rod and soluble boron worths are decreased.
- The seed-blanket configuration imparts better exploitation of the advantages of Th but has thermal hydraulic design issues.

Desirability of Establishing the Thorium Fuel Cycle for Nuclear Power Generation -3

- All the above advantages and disadvantages have been known for many years. It has been demonstated that the disadvantages can be designed around. The disadvantage of U-232 in the back-end of fuel cycle can be dealt with by using remotely-operated shielded facilities. Clearly the costs will be higher than those for the Uranium fuel cycle. A compensation of these costs may be possible by the greater efficiency of U-233 and the potential of higher burn-up with ThO_2. Most important issue would be the magnitude of the cost advantage of ThO_2 raw fuel over that of UO_2. It is interesting to compare the past situation with current situation in the nuclear power economy in the world.

Desirability of Establishing the Thorium Fuel Cycle for Nuclear Power Generation -4

Past Situation

In 1960s it did not make much economic sense to consider Thorium fuel for larger scale application, since there was the firm belief that the LWR and PHWR fuel will be reprocessed and that the Pu produced will be employed in fast reactors for an essentially unbounded nuclear growth scenario. In 1970s an argument for Th fuel cycle could be made due to the many orders for nuclear plants; however, by early 1980s the high growth scenario for nuclear power evaporated and the pressure on the U supply diminished.

Current Situation

- The LWRs are the ’King’, the main-stay of providing the large scale base-load power generation. The life of LWRs has been extended to 60-80 years. The need to assure the U supply for each plant upto 80 years could put pressure on the U supply. However the high growth scenario for LWRs is not credible today and the U fuel price may not escalate.

- Much Pu is available from dismantled weapons. The excess, in fact, is considered as a proliferation hazard.

- The U-Pu based fast reactor development is also considerably delayed.

A Recommendation

- I believe that it is prudent to develop an alternative to the U-Pu fuel cycle for future generations. However, it is not easy to make a case for commercial thorium fuel utilization only on the basis of economics. The uranium fuel prices are still much lower than 100$/pound or $220/Kg which is considered as the break-even price for introduction of alternate fuel cycle (fast reactors or thorium cycle)
- Thorium cycle does provide an efficient platform to burn and transmute Pu-239; in LWRs or fast reactors. The U-233 formed can not be easily used for weapons manufacture, since it has much U-232. Long burn-up operation of Pu-Th fuel could also burn the U-233 formed from thorium, thereby producing additional energy. Such an open fuel cycle could indeed be economically competitive with the open fuel cycle currently employed for LWRs. A “mission” of disposal of accumulated Pu could be justified for Th.
- In future when fuel cycles are closed, the U-233 made in Pu-Th fuelled LWRs or fast reactors could be extracted and employed for power generation in LWRs. This would require shielded facilities for reprocessing and re-fabrication, which would be expensive, but feasible.
- Research and pilot-scale developments for the Th cycle will need public financing. If governments assume that responsibility, commercial use could become feasible and desirable