THORIUM-BASED FUELS FOR PWRs

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Introduction

- Objectives of an Advanced Fuel Cycle
- Why Thorium-Based Fuel Cycles?
  - Benefits
  - Challenges
- Implementation Options and Issues
- Results of Recent Studies
- Conclusions
Objectives of an Advanced Fuel Cycle

- Reduce the long-term environmental burden of nuclear energy through more efficient management of waste materials
  - Remove/reduce transuranics (TRU) in the waste
  - More efficient utilization of long-term disposal space
  - Significantly reduce dose and radiotoxicity

- Enhance overall nuclear fuel cycle proliferation resistance via improved technologies for spent fuel management
  - Avoid generation of weapons usable materials
  - Improve inherent barriers and safeguards

- Enhance energy security by maximally extracting energy recoverable from spent fuel, avoiding uranium resource limitations
  - Extend nuclear fuel supply

- Continue competitive fuel cycle economics and excellent safety performance of the entire nuclear fuel cycle system
Recent Interest in Thorium-Based Fuel Cycles

- Recent examination of a broad spectrum of fuel cycles that can be used to meet advanced fuel cycle objectives has resulted in greater consideration of thorium-based fuels and fuel cycles
  - Thorium fuel could be used to replace (or supplement) uranium fuel provided an external source of fissile material is available to start the fuel cycle

- Significant body of work exists in U.S. on use of thorium in advanced fuel cycles; e.g.,
  - Major assessments during the U.S. NASAP and International INFCE programs
  - Study of utilization in thermal reactors (e.g., LWRs, HTR), MSRs, and fast reactors
  - Thorium fuel has been used in experimental and power reactors
  - Major demonstration effort at Shippingport in the 1970s/1980s
Why Thorium-based Fuel Cycle?

*Future Growth of Nuclear Power is Affected by Concerns Over:*

**Proliferation**  Thorium-Based Fuels have significantly reduced total plutonium production, and degraded/less favorable Pu isotopics for use in weapons. HOWEVER, U-233 is “weapons usable”, and represents an equivalent proliferation risk to plutonium in uranium-based fuels.

**Waste**  Thorium-Based Fuels are chemically more stable, and have higher radiation resistance than UOX → higher burnup potential; attractive option for once-through cycle (reduced production of transuranics can benefit repository performance; more durable and stable waste form, reduced waste per GWe, etc.)

Several experimental and prototype power reactors successfully operated in the 1950s to the 1980s using thorium fuel in high temperature reactors (HTRs, including Fort St. Vrain, European HTRs), LWRs (BORAX, Elk River, and Indian Point), and molten salt reactors. Focus of Indian nuclear program which includes design of advanced heavy water reactors (AHWRs) and generation of U-233 in FRs and ADS.

As a result, thorium-based fuels continue to be actively studied (international, IAEA, industry, etc.). for Pu management, waste/repository benefits, etc.
Thorium-Based Fuels: Benefits (IAEA-TECDOC-1450)

- Thorium is 3 to 4 times more abundant than uranium, and widely distributed in nature as an easily exploitable resource in many countries.
- Higher conversion (to 233U) is possible with 232Th than with 238U (to 239Pu). Thus, thorium is a better ‘fertile’ material than 238U in thermal reactors but thorium is inferior to depleted uranium as a ‘fertile’ material in fast reactor.
- For the ‘fissile’ 233U nuclei, the number of neutrons liberated per neutron absorbed (represented as \( \eta \)) is greater than 2.0 over a wide range of the thermal neutron spectrum; therefore the 232Th–233U fuel cycle can operate (and breed) with fast, epithermal or thermal spectra.
- Thorium dioxide is chemically more stable and has higher radiation resistance than uranium dioxide. The fission product release rate for ThO2–based fuels is one order of magnitude lower than that of UO2.
- ThO2 has favourable thermophysical properties \( \rightarrow \) expected to have better in–pile performance than UO2 and UO2–based mixed oxide (demonstrated at Indian Point).

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<tr>
<td>Melting temperature</td>
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<td>Coefficient of thermal expansion</td>
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<td>11.0 x10^-6 K^-1</td>
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Thorium-Based Fuels: Benefits (cont’d)

- ThO2 is relatively inert and does not oxidize unlike UO2, which oxidizes easily to U3O8 and UO3. Hence, long term interim storage and permanent disposal in repository of spent ThO2–based fuel are simpler without the problem of oxidation (degree of benefit depends on specifics of repository environment).
- Th–based fuels and fuel cycles have “intrinsic proliferation-resistance” due to the formation of 232U (daughters emit high energy gammas up to 2.7 MeV).
- More effective for incineration of WGPu or RGPu.
- In 232Th–233U fuel cycle, significantly reduced quantity of plutonium and long-lived Minor Actinides (MA: Np, Am and Cm) are formed as compared to the 238U–239Pu fuel cycle,
- ThO2 has only one oxidation state which is insoluble. While UO2 is insoluble in water, the higher oxidation states U3O8 and UO3 are. However in the mixed ThO2-UO2 UFla research showed that the ThO2 protected the UO2 from dissolution by water → operational and safety benefits.
Thorium-Based Fuels: Challenges (IAEA-TECDOC-1450)

- No naturally occurring fissile isotope → need external source to drive the system (critical or subcritical) to breed fissile U-233
- The melting point of ThO$_2$ is much higher compared to that of UO$_2$ → a much higher sintering temperature (>2,000°C) is required to produce high density ThO$_2$ and ThO$_2$–based mixed oxide fuels.
- ThO$_2$ and ThO$_2$–based mixed oxide fuels are relatively more difficult to process.
- The irradiated Th or Th–based fuels contain significant amount of 232U. As a result, there is significant buildup of radiation dose with storage of spent Th–based fuel or separated 233U, necessitating remote and automated reprocessing and refabrication in heavily shielded hot cells and increase in the cost of fuel cycle activities.
- Formation of 233Pa which has a relatively longer half-life (~27 days) as compared to 239Np (2.35 days) in the uranium fuel cycle requires longer cooling time of at least one year for completing the decay of 233Pa to 233U. Affects processing.
- The three stream process of separation of uranium, plutonium and thorium from spent (Th, Pu)O$_2$ fuel, though viable, is yet to be developed.
- The database and experience of thorium fuels and thorium fuel cycles are very limited, as compared to UO$_2$ and (U, Pu)O$_2$ fuels, and need to be augmented before large investments are made for commercial utilization of thorium fuels and fuel cycles.
Implementation Options

- Compatible with thermal, epithermal and fast spectrum systems
- No fissile isotope in natural thorium → fissile material/neutron source needed to “drive” system

**Reactor Systems**
- Once-Through: <20 w/o U-235, WGPu, + Thorium
- Recycle: <12 w/o U-233, RGPu, TRU, MA + Thorium
- Spectrum of “synergistic cycles” (uranium/thorium cycles)
- Implementation options include homogeneous (all fuel rods in core contain mixture of fissile + thorium) and heterogeneous (thorium is present only in some portion of the fuel rods/assemblies)

**Driven Systems (ADS, FFH, etc.)**
- Fissile material in blanket region can include same as above, or “breed and burn” using neutrons from accelerator or fusion source, and/or combined with sub-critical
- Breed fissile material, burn TRU., generate power (FFH LIFE concept from LLNL)
- “Energy Amplifier”, waste burning (ADS)
Issues

Source of U-233 to sustain “pure” U-233/Th cycle
- Initial studies have shown that systems based on current PWRs cannot produce sufficient U-233 to be self-sustaining
- Options considered under Bettis Atomic Power Lab (BAPL) Advanced Water Breeder Applications program, included:
  - Pre-Breeder based on “commercial” PWR to provide initial source of U-233
  - Breeder reactor based on scale-up of Light Water Breeder Reactor (LWBR)
- Molten Salt Reactor, fast reactor, or accelerator or fusion based “Driven System” could also serve as source of U-233 (support ratio?)
- For thermal spectrum systems, configurations different from current commercial LWRs are required for breeding (e.g., very tight lattice, reduced power density)

Proliferation Risk
- U-233 is similar to Pu-239 in attractiveness as “weapons usable material”
- High-energy gamma (2.6 MeV) from U-232/daughter chain provides radiological/dose deterrent but effectiveness depends on quantities, and adversary
- Sufficient addition of U-238 to “denature” bred U-233 (<12 w/o) required to make material not usable in a weapon
**Issues (cont’d)**

- **Waste characteristics (decay heat, curies)**
  - Depend strongly on details of implementation scenario (reactor type and configuration, fissile material, etc.)
  - Similar to uranium-based fuel for initial ~100-yrs when fission products dominate
  - Higher than for UOX fuel at 100,000-yrs. due to radon, etc.

- **Processing of thorium-based fuels**
  - THOREX process demonstrated and used successfully; **However**, process is generally more complex than for processing of UOX fuels
  - Limited experience, so commercial attractiveness TBD
  - Multi-element extraction which is desirable for certain applications has not been considered or explored

- .....
“Recent” Work on Thorium-Based Fuel Cycles

IAEA and Other European


U.S. NERI and University

- INEEL NERI (1999): Homogeneous and micro-heterogeneous (fuel rod level) implementation in PWRs
- BNL NERI (2000): Heterogeneous implementation in PWRs.
- BNL NERI (1999): Tight-hex-lattice BWR
- Focus on once-through, as replacement for UOX in conventional reactors.
- Studies continue
“Recent” Work on Thorium-Based Fuel Cycles (cont’d)

**DOE-IPP and MD**

- BNL-Kurchatov-Radkowsky Thorium Power Corp/Thorium Power project was started in 1996 under the DOE Initiatives for Proliferation Prevention (IPP) program – BNL was the Project Manager until the DOE funding ended. A lot of work was done, including reactor design, safety analysis, prototypic fabrication, irradiation testing of U-Zr and (Th,U)O₂ fuel samples in IR-8 reactor, and T-H experiments.

- ORNL-Kurchatov-Westinghouse: essentially continuation of above looking at WG-Pu burning variant in response to Congressional directive performed under Materials Disposition program.

**DOE - AFCI/FCR&D**

- Multiple thorium cycle scenarios/options evaluated under FCR&D Systems Analysis/Engineering Campaigns
- Includes thermal and fast reactors, driven systems, once-through & closed cycle options

**Other**

- Industry (Lightbridge Inc., Thor Energy, Thorium Energy, Thorenco, ...)
- ADS and FFH communities (Energy Amplifier, and descendants; LIFE; …)
- International (India, China, …)
Once-Through PWR Thorium-Based Fuel Cycle

- Constrained to be retrofittable in existing PWRs
- <20 w/o U-235 (bred U-233 denatured)
- Heterogeneous schemes “better” than homogeneous:
  - Can more easily optimize physics/thermal-hydraulics of seed and blanket (S/B) zones
  - Separate fuel management approach for each region → easier to maximize advantages of thorium
- Several benefits vs. UOX:
  - Reduced Pu production (~3x)
  - “Degraded” Pu isotopics
  - Reduced mass and volume of waste per GWe-yr
- Challenges
  - High power production in seed (T-H)
  - High burnup in both seed and blanket needed to realize benefits (fuel performance)
- Comparable safety, fuel cycle costs,…
- Collaboration with MIT and Ben-Gurion University
- Complemented by work in Russia under DOE/IPP program and Thorium Power
Brief History of VVERT Project

OBJECTIVE: To develop a variant of the seed-blanket unit (SBU) for implementation in a VVER-1000, and perform initial testing

- Project funded under the DOE Initiatives for Proliferation Prevention (IPP) program and Radkowsky Thorium Power, Corp. (aka Thorium Power/Lightbridge Inc.)
- Managed by Russian Research Center-”Kurchatov Institute”(RRC-KI), and included several institutes (e.g., PJSC "Electrostal", NPO "LUCH", etc.)
- Major tasks included:
  - Development of VVERT design based on SBU concept and confirmation of performance and safety characteristics (with participation of GAN).
  - Identify and perform initial thermal-hydraulic experiments with view to support licensing and LTA program.
  - Identify and perform initial fabrication and testing for seed and blanket fuel options
  - Perform an initial assessment of the costs and economic potential

These activities confirmed viability of concept and that there were no “showstoppers”

Follow-on MD Program: In summary, based on the overall assessment of the risks and potential benefits of pursuing the RTPI concept, it is Westinghouse’s opinion that proceeding to the LTA stage is prudent. From the review that we have performed to date, it appears that the technology is well founded and has a good prospect for success based on our previous US experience and Russian experience with metal fuels. The true economics of this plutonium disposition approach compared to the MOX approach are likely to be favorable if existing facilities can be used with only the modifications necessary to accomplish the fabrication of the plutonium-thorium seed and blanket fuel.
VVERT Seed Blanket Unit Design

<table>
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<tr>
<th>Parameter</th>
<th>VVERT1000-Seed Blanket Unit (SBU)</th>
<th>Reference VVER-1000</th>
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<tr>
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<td>Seed Region</td>
<td>Blanket Region</td>
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<tr>
<td>Fuel Rod geometry</td>
<td>Three-petal</td>
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<tr>
<td>Number of Rods per assembly</td>
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<tr>
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<td>Fuel Pellet central hole Diameter, cm</td>
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<td>Clad inside diameter, cm</td>
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<td>(U-Th)O₂</td>
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<td>9.5 – ThO₂</td>
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<td>10.4</td>
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<tr>
<td>Cladding Material</td>
<td>Zr + 1% Nb</td>
<td>Zr + 1% Nb</td>
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![Diagram of VVERT Seed Blanket Unit](image-url)
Fuel Samples Manufactured At Msz Electrostral
Facility For Full-scale Thermal-hydraulic Tests (VVERT & PWRT)
Fuel Sample Irradiation Tests At IR-8
Example Scenarios


- Once-thru “equilibrium” TRU-Th oxide fuel cycle assuming an infinite source of TRU (Np, Pu, Am) based on UNF produced by an advanced light water reactor (ALWR) with a discharge burnup of 50 GWd/T, and cooled for five years prior to separating the TRU from the remaining UNF, followed by a 2-year period for fabrication and transport prior to insertion into the reactor.

- “Self-Re-cycle” U-TRU-Th oxide fuel cycle starting with the TRU-Th as in Scenario (2) and then recycling the discharged U and TRU (Np, Pu, Am), with additional makeup as needed from ALWR discharge.
U-233-Th Fuel Cycle

- Essentially no TRU produced
- No “denaturing” so rely on U-232 content ~3000 ppm for “proliferation resistance”
- Radioactivity (and decay heat) not significantly different from that of conventional UOX with significant increase at ~10^5 yrs
- Amount of U-233 generated not enough to sustain self-recycle → (Pre)Breeder required
TRU-Th Fuel Cycle

- Total Pu is reduced by 40% with Th matrix vs. 17% with U
- Reduction in Pu-239 is twice as large with Th matrix
- Radioactivity and decay heat vs. time are similar, with characteristic “bump” at ~10^5 yrs

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Kg/assembly
U-TRU-Th Fuel Re-Cycle
Summary

- Thorium-based fuels and fuel cycles have attractive properties and continue to be pursued:
  - Reduced production of Pu and minor actinides
  - Degraded Pu isotopics and “enhanced self-protection”
  - Enhanced sustainability/resources
  - Improved in-core and ex-core characteristics

- Clear benefits for burning plutonium relative to MOX, but “spent fuel standard” is deemed adequate for disposition of WGPu – however, degree desired depends on objectives

- More advanced implementation options come with both benefits and challenges:
  - Greatest benefits relative to conventional UOX appear to require transition to U-233 (eliminate need for enrichment, significant reduction in TRU production, etc.)
  - Will likely require “unconventional” reactor designs to optimize benefits
  - Proliferation concerns with U-233 may require denaturing which minimizes benefits
  - Recycling and fabrication of new fuel containing U-233 introduce challenges for processing and remote operations
  - ....