Indian approach for radioactive waste management

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REPUBLIC OF INDIA

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The Three Stages of the Indian Nuclear Program

Stage 1: Power generation primarily by PHWR
Building fissile inventory for stage 2

Stage 2: Expanding power program
Building U\(^{233}\) inventory

Stage 3: Thorium utilisation for Sustainable power program

PHWR

U\(^{233}\) fuelled research reactor KAMINI

U\(^{233}\) fueled PHWRs

Pu Fueled Fast Breeders

Pu\(^{233}\) Fueled Reactors

Homi Jehangir Bhabha 1909-1966
India has decades of experience in exercising closed fuel cycle involving reprocessing, conditioning and recycling of fissile material.
Indian Nuclear Fuel Cycle : Today

U (natural) → UOX fuel → PHWR → Purex

- Volume Reduction
- Near Surface Disposal
- Non-α waste (L&IL)

Purex → Heavy metal

- α Bearing HLW

Reprocessing Facility

Reprocessing of carbide fuel

- Mixed Carbide fuel for Fast breeder test reactor
- Mixed Oxide fuel for Prototype Fast Breeder Reactor

Purex → Vitrification Facility

- Vitrification
- Interim Storage
- α Bearing HLW

Vitrification Facility

Vitrified Waste Storage Facility
National Guidelines of Waste Management

- Protection of human health, environment and future generations from Nuclear Radiations.

- No waste is discharged or disposed till it is exempted or cleared by the regulatory body abiding ALARA.

- Disposal of Low & ILW in Near Surface Disposal Facilities.

- Co-Location of NSDF with Nuclear facilities - Institutional control of about 300 yrs.

- Spent Fuel Used as resource Material – Recycle of Fissile / fertile Material (Reprocess Condition and Recycle principle).


- Partitioning of waste for separation of minor actinides and useful fission products.

- Spent radiation sources are managed by Waste Management Division, BARC at no cost basis.
Basic philosophy for management of radioactive waste

**OBJECTIVES**

*Protection of*

- HUMAN HEALTH
- ENVIRONMENT
- FUTURE GENERATION

**BASIC TENETS**

- DELAY AND DECAY
  *(short half-life, $\beta-\gamma$ active)*
- DILUTE AND DISPERSE
  *(low activity)*
- CONCENTRATE AND CONTAIN
  *(high activity)*
- REDUCE, RECYCLE AND REUSE
- PARTITION AND TRANSMUTATE
## Classification of radioactive wastes

<table>
<thead>
<tr>
<th>Category</th>
<th>Solid</th>
<th>Liquid</th>
<th>Gaseous</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Surface Dose (mGy/hr)</td>
<td>Activity level (Bq/M³)</td>
<td>Activity level (Bq/M³)</td>
</tr>
<tr>
<td>I</td>
<td>&lt;2</td>
<td>&lt; 3.7×10⁴</td>
<td>&lt; 3.7</td>
</tr>
<tr>
<td>II</td>
<td>2-20</td>
<td>3.7×10⁴ to 3.7×10⁷</td>
<td>3.7 to 3.7×10⁴</td>
</tr>
<tr>
<td>III</td>
<td>&gt;20</td>
<td>3.7×10⁷ to 3.7×10⁹</td>
<td>&gt; 3.7×10⁴</td>
</tr>
<tr>
<td>IV</td>
<td>Alpha Bearing</td>
<td>3.7×10⁹ to 3.7×10¹⁴</td>
<td>-</td>
</tr>
<tr>
<td>V</td>
<td>-</td>
<td>&gt; 3.7×10¹⁴</td>
<td>-</td>
</tr>
</tbody>
</table>
Summary of Indian radioactive waste management practices

Characterization

- LL: Liquid
- IL: Solid
- HL: Gaseous

Treatment

- Liquid Waste:
  - Chemical Treatment
  - Ion Exchange
  - Reverse Osmosis
  - Evaporation
- Solid Waste:
  - Compaction
  - Incineration
  - Size Fragmentation
  - Repackaging
- Gaseous Waste:
  - Scrubbing
  - Adsorption/Absorption
  - Prefiltration
  - High Efficiency Filtration

Conditioning

- Cementation
- Polymerisation
- Vitrification

Interim Storage

- Alpha Contaminated Waste
- Wastes requiring treatment / conditioning in future
- Vitrified waste for cooling pending disposal

Disposal

- LIL Waste Short lived
- HLW & Long Lived Waste
  - Earth/Stone lined trenches
  - Reinforced Concrete trenches
  - Tile Holes
  - Deep Geological Disposal
  - In waiting

Environmental Monitoring/Control

- Monitoring of water, soil, vegetation, near waste management facility.
- Monitoring of environment near nuclear facility.
- Institutional control of near surface disposal facility for 300 years.
## Generation of gaseous waste

<table>
<thead>
<tr>
<th>SOURCE</th>
<th>RADIONUCLIDE</th>
</tr>
</thead>
<tbody>
<tr>
<td>MINING</td>
<td>RADON AND ITS DAUGHTER PRODUCTS</td>
</tr>
<tr>
<td>NUCLEAR REACTOR OPERATION</td>
<td>IODINE- 131</td>
</tr>
<tr>
<td>SPENT FUEL REPROCESSING</td>
<td>KRYPTON-85</td>
</tr>
<tr>
<td>VITRIFICATION OF HIGH LEVEL RADIOACTIVE WASTE</td>
<td>CESIUM-137, RUTHENIUM-106</td>
</tr>
</tbody>
</table>

In addition, PARTICULATES are always associated with the gaseous stream in any of these facilities. These particulates are removed by filtration using pre-filters and HEPA filters.
MANAGEMENT OF GASEOUS WASTE

- Scrubbing of volatile contaminants
- Adsorption
- Condensation
- Chemical impregnated activated charcoal
- HEPA Filters
  (High Efficiency Particulate Air Filters made of micro-glass fibers)
- Dilute and Disperse
- Delay and Decay
Treatment processes for low level solid wastes

**COMPACTION**
(plastic & rubber)

**COMBUSTION**
(cellulosic)

**MELT DENSIFICATION**
(Polythene)

THE NON-TREATABLE WASTES ARE DIRECTLY DISPOSED.
Options for disposal of radioactive solid waste

- **NEAR SURFACE DISPOSAL**
  - DEPTH LIMITED TO 10 M
  - DISPOSAL OF LOW AND INTERMEDIATE LEVEL WASTE

- **DEEP GEOLOGICAL DISPOSAL**
  - DEPTH 500-600 M
  - DISPOSAL OF HIGH LEVEL RADIOACTIVE WASTE

Brick Walled/Stone Lined Trenches (BWT/SLT)
Reinforced Cement Concrete Trenches (RCT)
Tile hole (TH)
Radioactive liquid waste have been traditionally categorized into three categories depending on the concentration of radionuclides

- **Low Level Waste (LLW)**
  - NEAR NEUTRAL
    - Vol. % - 90
    - Activity % - 0.5-1
  - ALKALINE
    - Vol % - 7
    - Activity % - 1-2
- **Intermediate Level Waste (ILW)**
- **High Level Waste (HLW)**
  - ACIDIC
    - Vol % - 3
    - Activity % - 98-99
Low level liquid waste (LLW) - treatment

- **Chemical Precipitation**
  - Coagulation, flocculations

- **Ion-exchange**
  - PU foam coated with Cu-Ferrocyanide and Hydrous Manganese Oxide as exchanger

- **Reverse Osmosis**
  - For waste containing low solids
  - Disc type and Spiral Wound type membrane

*Treatment of LLW to achieve near zero discharge activity discharge to environment*
Selective ion exchange process for treatment of intermediate level waste

- Resorcinol Formaldehyde Polycondensate Resin (RFPR) for removal of Cs-137
  - Indigenously developed
  - Used in loading-elution cycles
  - In-house facility for production

- Industrial scale plants treat ILW at Trombay and Tarapur using RFPR
  - 3000 M³ treated
  - Emptying of legacy waste storage tanks
  - Treatment of De-clad waste

- Wealth from waste
  - Recovery of Cs-137 in KCl quantities for use as radiation source (e.g., blood irradiator)

**Development Targets: Complete elimination of ILW**

[Diagram showing IX Facility, Trombay]
Management of High Level Waste: 3-Stage Program

- Canister Welding
- Shielded Transport Cask
- Air cooled Storage
- Deep Geological Repository

- HLW from Reprocessing
- Immobilization
- Interim Storage
- Deep Disposal

HLW from Reprocessing
HLW MANAGEMENT PLANTS

WIP, Tarapur

WIP, Trombay

AVS, Tarapur

Bhabha Atomic Research Centre
www.barc.gov.in
Three generations of Melters

Induction Heated Metallic Melter
- Capable of handling variation in waste characteristics
- Ease of operation and handling
- Easy decommissioning

Joule Heated Ceramic Melter
- High temperature
- High throughput
- Continuous operation

Cold Crucible Induction Melter
- Higher temperatures
- Feasibility for future matrices
- Expected long melter life
Objectives:

- To allow dissipation of decay heat
- Continuous surveillance of VWP

Salient Features:

- Cooling provided
- Elaborate temperature measurements for air, overpack surface, concrete, air borne activity
Concept of geological disposal

**SELECTION OF GEOLOGICAL MEDIA**

- Granitic formation lying in tectonically stable zones.
- Massive uniform and large extent in three dimensions.
- Minimum fracture zones, joints and intrusions.
- Without ground water and away from major surface water bodies.
- Away from the known mineral deposits.
- Away from dense population areas, the tourist spots and ecologically important places.
Reduction of Radiotoxicity

Waste after reprocessing with actinide partitioning

Waste after reprocessing without partitioning

Radiotoxicity of Uranium Ore
Impact on Geological Disposal Facility

Vitrified Waste for Disposal

Geological Disposal Capacity (Te)

Without Partitioning

With Partitioning
WEALTH FROM NUCLEAR WASTE

Recovery of valuable fission products for societal benefits

Cesium-137
- Blood Irradiation

Ruthenium-106
- Irradiation source and for cancer treatment (under development)

Strontium-90
- Radio – pharmaceutical & Power pack

Americium-241
- Neutron source
- Space application
Adoption of pre-treatment with Partitioning of waste using Novel solvents

- In house Development
- Commercial Production
- Plant Scale Deployment
- Ligand Modelling

FABRICATION OF NOVEL SOLVENTS IN LARGER AMOUNT ENABLED THE PARTITIONING OF WASTE
Cs-137 Pencil Irradiator (Recovery of Cesium-137 from HLW)

Processes for separation of Cs-137 from HLW
• Available in nuclear waste in large quantities
• Solvent extraction method (calix-arene based solvent system)

Vitrified form of Cs-137
• Serves as a better radiation source with enhanced chemical durability and longer service life

<table>
<thead>
<tr>
<th>Characteristics</th>
<th>Co-60</th>
<th>Cs-137</th>
</tr>
</thead>
<tbody>
<tr>
<td>Half Life</td>
<td>5.27 Years</td>
<td>30 Years</td>
</tr>
<tr>
<td>Specific Activity</td>
<td>25-300 Ci/g</td>
<td>5-15 Ci/g for glass matrix</td>
</tr>
</tbody>
</table>

Induction heated metallic melter for vitrification
Remote welding and decontamination of pencil
Second Stage of Indian Nuclear Program

FBTR operating for 3 decades @ Kalpakkam
PFBR @ advance stage of construction

Fuel Composition: MOx
Fissile content: 25%
Expected burn up: 100GWD/T
HLW management strategies for fast reactor wastes

- Short cooled and higher burn ups leading to higher specific activity.
- Delay of HLW processing for six years to allow decay of short lived isotopes.
- Comparatively higher volumes of waste.
- Pretreatment of waste for further recovery of useful elements.
- Need for high temperature glass matrix.
- Vitrification of HLW using by Joule Melter/ CCIM.
- Need of Exhaustive off-gas treatment system.
- Secondary waste: Returned for concentration.
- VWP Canister Storage at centralized VWSF, Kalpakkam.
Thoria spent fuel reprocessing- challenges

– Head-End
  ▪ Flouride induced dissolution
    ▪ Enhanced dissolution of Thoria Vs corrosion **attack due to flouride**
    ▪ Optimization of Al(NO$_3$)$_3$ addition- **Impact on Waste management**

– Extraction
  ▪ Three component system, Third phase formation

– Partitioning
  ▪ **Chemical reductants** in place of U$^{+4}$ to avoid contamination in U-233

– Remote product handling
  ▪ Handling & storage of Th-232
  ▪ **Presence of U-232** with U-233 and its cleanup
Challenges in Thorium Fuel Cycle

- Dissolution of Sintered ThO$_2$ fuel
- Addition of Fluoride in during dissolution
  
  \[12 \text{M HNO}_3 + 0.1 \text{M Al(NO}_3)_3 + 0.03 \text{M NaF}\]

  Leading to Corrosion problem
- Radiological problems associated with $^{232}$U
- Waste management problems
  
  Th, Al & F$^-$
  
  Decreased Waste loading
  
  Viscosity hence Pouring temperature
  
  Corrosion problems
- Lesser Experience
  
  ✓ Reprocessing: Dissolution
  
  ✓ Waste management: Thorium Lean Raffinate
Summary of waste management experience of India

- Few decades of safe and rich operating experience at various sites.
- Conditioned Products are comparable to best international standards.
- Adoption of partitioning makes India take the lead role.
- Separation of fission products to define radioactive waste as wealth.
- Attempts towards minimization of long term radiotoxicity due to actinides.
Thank You...
<table>
<thead>
<tr>
<th>PHWR(500Mwe)</th>
<th>FBR(500 Mwe)</th>
<th>PWR(500 Mwe Eq.)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>6.7 GWD/Te</strong></td>
<td>FSA: 100 GWD/Te RSA: 4-5 GWD/Te</td>
<td>50 GWd/Te</td>
</tr>
<tr>
<td><strong>80 Te fuel/Yr</strong></td>
<td>15 Te fuel/yr</td>
<td>11 Te fuel/year</td>
</tr>
<tr>
<td><strong>Specific activity 100 Ci/kg of fuel</strong></td>
<td>Specific activity Thousands of Ci/kg of fuel.</td>
<td>Few hundreds of Ci/kg of fuel</td>
</tr>
<tr>
<td><strong>1 Cu.M of HLW/Te</strong></td>
<td>10 Cu.M of HLW/Te</td>
<td>1 Cu.M of HLW/Te</td>
</tr>
<tr>
<td><strong>1 canister equivalent to 6 Cu.M of HLW (considering 0.6 M Ci loading, 25% W.O, 1.8 KW/canister)</strong></td>
<td>1 canister equivalent to 11Cu.M of HLW (0.6 Mci loading, 14% W.O, 1.8 KW/canister,)</td>
<td>1 canister equivalent to 1 Cu.M of HLW (considering 0.6 M Ci loading, 5 % W.O, 1.8 KW/canister)</td>
</tr>
<tr>
<td><strong>13 Canister/yr</strong></td>
<td>14 canister/yr</td>
<td>15 canister/yr</td>
</tr>
</tbody>
</table>
## Waste management facilities in India

<table>
<thead>
<tr>
<th>Site</th>
<th>Location</th>
<th>Year</th>
<th>Nuclear facilities</th>
</tr>
</thead>
<tbody>
<tr>
<td>COASTAL</td>
<td>Trombay</td>
<td>1956</td>
<td>Research Reactors, Fuel Fabrication, Fuel Reprocessing, Research Labs, Isotope Production, WIP</td>
</tr>
<tr>
<td>COASTAL</td>
<td>Tarapur</td>
<td>1969</td>
<td>BWR(2x160 MWe), PHWR(2x540 MWe), Fuel Reprocessing, Fuel Fabrication, WIP</td>
</tr>
<tr>
<td>COASTAL</td>
<td>Kalpakkam</td>
<td>1984</td>
<td>PHWR (2 X 220 MWe), Fuel Reprocessing, Research Labs, Research Reactors (FBTR), PFBR and WIP under construction</td>
</tr>
<tr>
<td>INLAND</td>
<td>Rajasthan</td>
<td>1972</td>
<td>PHWR (6 X 220 MWe), Isotope facility</td>
</tr>
<tr>
<td></td>
<td>Narora</td>
<td>1989</td>
<td>PHWR (2 X 220 MWe)</td>
</tr>
<tr>
<td></td>
<td>Kakrapar</td>
<td>1993</td>
<td>PHWR (2 X 220 MWe)</td>
</tr>
<tr>
<td></td>
<td>Kaiga</td>
<td>2000</td>
<td>PHWR (4 X 220 MWe)</td>
</tr>
<tr>
<td></td>
<td>Kudunkulam (under construction)</td>
<td>2014</td>
<td>LWR (1000 MWe)</td>
</tr>
</tbody>
</table>
### Comparison with Natural U fuel cycle

<table>
<thead>
<tr>
<th>Reprocessing</th>
<th>Wastes from Fast Reactor</th>
<th>Wastes from Natural uranium Reactor</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Short cooled &amp; High burn up fuel:</td>
<td>Long cooled &amp; low burn up fuel:</td>
</tr>
<tr>
<td></td>
<td>• High specific activity and high PGM</td>
<td>• Low specific activity and Low PGM</td>
</tr>
<tr>
<td></td>
<td>• Solvent degradation more: required centrifugal extractor</td>
<td>• Solvent degradation less</td>
</tr>
<tr>
<td></td>
<td>Fissile material more:</td>
<td>Fissile material less:</td>
</tr>
<tr>
<td></td>
<td>• Criticality aspect more: more annulus tank &amp; liquid poison in dissolver required</td>
<td>• Criticality aspect less: less no. annulus tank &amp; no liquid poison in dissolver</td>
</tr>
<tr>
<td></td>
<td>• more number of solvent extraction cycle for required DF</td>
<td>• less number of solvent extraction cycle for required DF</td>
</tr>
</tbody>
</table>
# Comparison with Natural U fuel cycle

<table>
<thead>
<tr>
<th></th>
<th>FRFCF</th>
<th>PHWR</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Waste Management</strong></td>
<td>➢ Waste Generation</td>
<td>➢ Waste Generation</td>
</tr>
<tr>
<td></td>
<td>• HLW: High (10 Cu.M/Te)</td>
<td>• HLW:Low (1 Cu.M/Te)</td>
</tr>
<tr>
<td></td>
<td>• ILW: High (5 Cu.M/Te)</td>
<td>• ILW:Low (2 Cu.M/Te)</td>
</tr>
<tr>
<td></td>
<td>• OLW: Very High (5 Cu.M/Te)</td>
<td>• OLW: Low (0.5 Cu.M/Te)</td>
</tr>
<tr>
<td></td>
<td>➢ HLW Characteristics</td>
<td>➢ HLW Characteristics</td>
</tr>
<tr>
<td></td>
<td>• Actinides: High</td>
<td>• Actinides: Low</td>
</tr>
<tr>
<td></td>
<td>• Platinum group Metal: High</td>
<td>• Platinum group Metal: Low</td>
</tr>
<tr>
<td></td>
<td>• Inactive contents: Low TDS, Gd-liquid poison</td>
<td>• Inactive contents: High TDS, No-liquid poison</td>
</tr>
<tr>
<td></td>
<td>• Lesser Waste loading</td>
<td></td>
</tr>
<tr>
<td></td>
<td>➢ Other Difference</td>
<td>➢ Other Difference</td>
</tr>
<tr>
<td></td>
<td>• SS Hulls</td>
<td>• Zircaloy Hulls</td>
</tr>
<tr>
<td></td>
<td>• High volume of alpha solid waste</td>
<td>• Less volume of alpha solid waste</td>
</tr>
<tr>
<td></td>
<td>• High volume of reactor hardware component.</td>
<td>• Less volume of reactor hardware component.</td>
</tr>
</tbody>
</table>
Summary of Indian experience in thorium cycle

- Thoria fuel bundles loaded in PHWRs
- Testing of (Th-Pu)MOX pins of BWR/PHWR type
- Post Irradiation Examination

- Use of thoria in PHWRs
- (Th-Pu)MOX Test Pins
- Thorium irradiation in research reactor
- Irradiated fuel reprocessing and fuel fabrication
- KAMINI research reactor: \( ^{233}\text{U-Al} \) Fuel
- Thorium irradiation in CIRUS, Dhruva & FBTR
- Reprocessing & fabrication of \( ^{233}\text{U} \) based fuel
- PURNIMA II & KAMINI test reactors