

Week 3 - Day 1

Nuclear Fuel Cycle

Lecture 1 – Introduction to the Nuclear Fuel Cycle

Lecture 2 – Uranium Mining, Milling, and Conversion

Lecture 3 – Uranium Enrichment

Introduction to the Nuclear Fuel Cycle Lecture 1

GNEII Pilot Course

February – May 2011

Khalifa University of Science, Technology and Research,

Abu Dhabi, United Arab Emirates

(William Charlton)

Texas A&M University, USA

SAND 2010_____

Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

Introduction to the Nuclear Fuel Cycle

- **Main Teaching Points**

- Nuclear Fuel Cycle
- Fuel Cycle Steps
- Closed and Open Fuel Cycle
- Nuclear Fuel
- Uranium and Thorium Ores and Supply
- Nuclear Reactor & Spent Fuel
- Spent Fuel Storage and Management

Introduction to the Nuclear Fuel Cycle

- **Key Terms** – nuclear fuel, fuel cycle (open and closed), fuel cycle (front end and back end), uranium ores, thorium ores, uranium reserves, yellowcake, reactor, fission products, spent fuel management
- **Key Concepts** – Differences between open and closed nuclear fuel cycles; Essential difference between using uranium and thorium as nuclear fuel; How to identify a fuel cycle; Locations of the largest uranium deposits; Spent nuclear fuel characteristics; Ways of spent nuclear fuel management

Introduction to the Nuclear Fuel Cycle

- **Desired Student Outcomes**

- Describe how nuclear fuel cycles are categorized
- Explain the steps involved in the nuclear fuel cycle
- Identify the advantages and disadvantages of closed fuel cycle
- Identify the global distribution and names of the principle uranium and thorium bearing ores
- Compare the usage of uranium and thorium as nuclear fuel
- Discuss what happens to nuclear fuel in reactor
- Recognize methods and pathways for spent fuel management

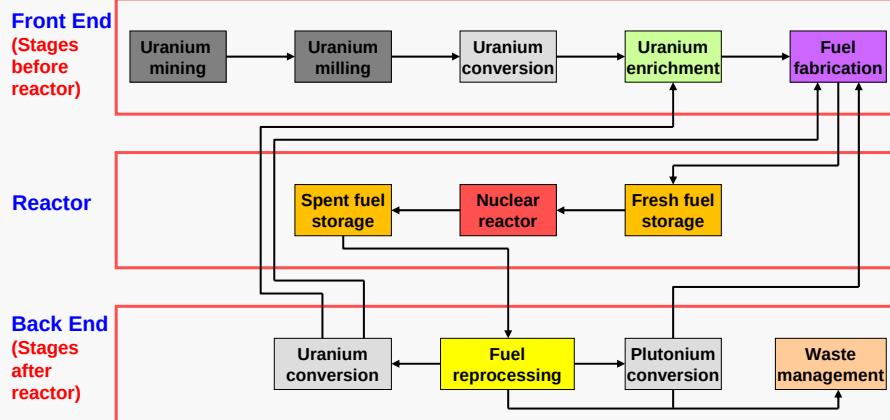
Assumptions and Prerequisites

- Successful completion of GNEII weeks one and two
- Bachelor's degree
- College level math

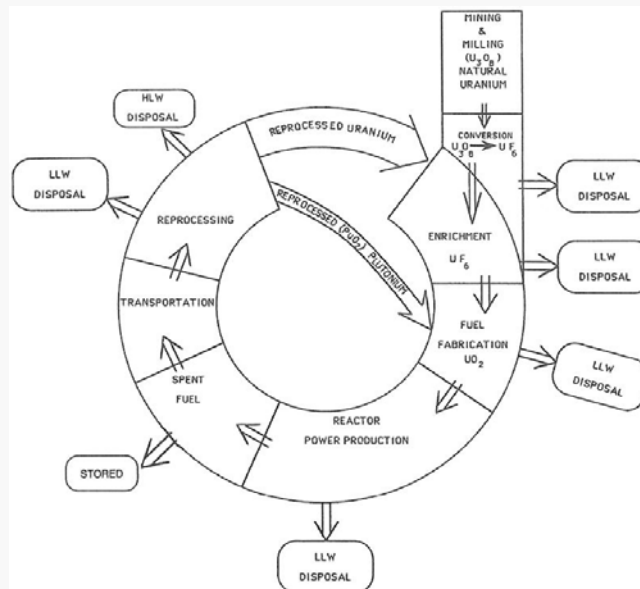
Definition of Nuclear Fuel Cycle (NFC)

The path followed by nuclear material (in its various states) during its use, through a system of interconnected nuclear facilities (from mining of ore to disposal of the final waste).

Schematic of Generic Commercial NFC



Nuclear Fuel Cycle

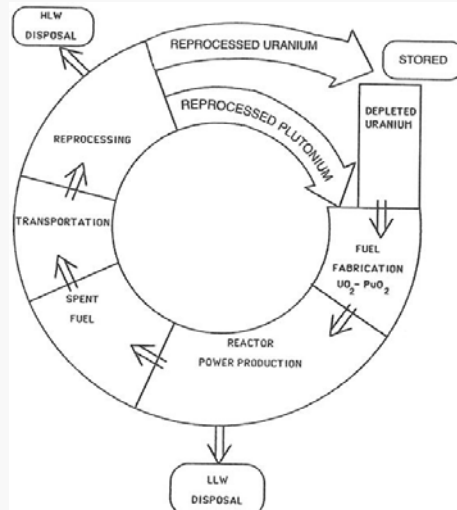


R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management*, 2nd Edition, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

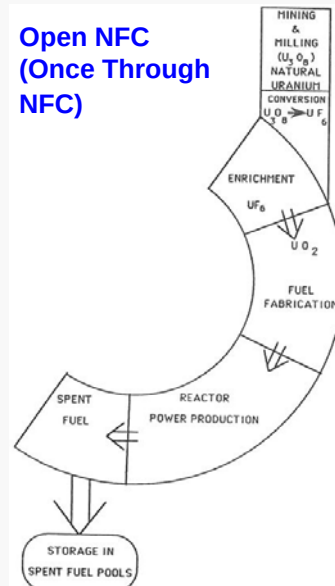
Two Types of Nuclear Fuel Cycles

Closed NFC

- Better uranium utilization
- Proliferation concerns



Open NFC (Once Through NFC)



R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management*, 2nd Edition, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

Nuclear Fuel - 1

- Most nuclear reactors use **uranium** (either natural or enriched) as the fuel
 - So we need to learn about uranium mining and milling
- Most commercial operating power reactors are **thermal reactors**
- Most thermal power reactors use uranium fuel in its oxide form (UO_2)

Uranium Isotopics

- Natural uranium consists of three isotopes:
 - ^{234}U (0.0055%), ^{235}U (0.72%) and ^{238}U (99.2745%)
- ^{235}U is the most useful isotope for nuclear power production in **thermal reactors**
 - Natural uranium is used in heavy-water moderated nuclear power reactors
 - Example: CANDU reactors
 - Natural uranium can also be used in graphite moderated nuclear power reactors
 - Example: MAGNOX reactors
 - Enriched uranium (enriched in ^{235}U to about 5%) is used in Light Water Reactors (LWRs)
 - Examples: Pressurized Water Reactors (PWRs) and Boiling Water Reactors (BWRs)

Other Nuclear Fuels

- ^{235}U , ^{233}U , and ^{239}Pu are the major isotopes which undergoes nuclear fission in reactors to produce nuclear power
- ^{233}U can be produced by introducing thorium in nuclear reactors
 - no commercial operating reactors currently use this fuel, except one or two research reactors
- Similarly, plutonium (^{239}Pu) is a byproduct in spent uranium fuel

Sources of Uranium

- Uranium minerals and ores
 - Pitchblende and Uraninite
 - 50% to 80% U_3O_8 content
 - chief constituent of all high grade uranium and largest part of all uranium produced world wide
 - Carnotite, Torbernite, Tyuyamunite, Autunite, Uranophane, Brannerite
 - 45% to 60% U_3O_8 content
 - Davidite, Samarskite, Euxenite
 - 1% to 18% U_3O_8 content
- All minerals containing uranium may not be called a mineral if uranium cannot be economically extracted from it

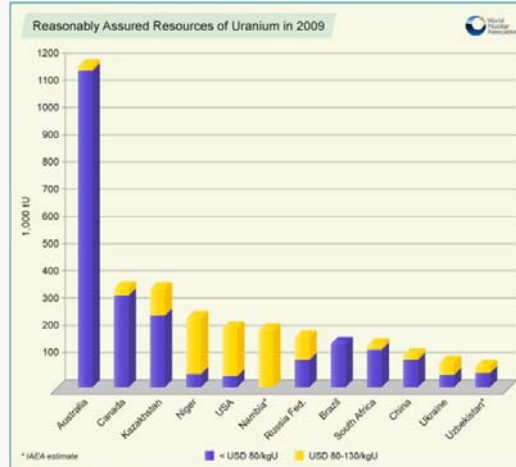
Uranium Ore Quality

Very high-grade ore (Canada)- 20% U	200,000 ppm U
High-grade ore - 2% U	20,000 ppm U
Low-grade ore - 0.1% U	1,000 ppm U
Very low-grade ore (Namibia) – 0.01 % U	100 ppm U
Granite	4-5 ppm U
Sedimentary rock	2 ppm U
Earth's continental crust (av)	2.8 ppm U
Seawater	0.003 ppm U

<http://www.world-nuclear.org/info/inf75.html> (updated December 2010)

Supply of Uranium

- The world's present measured resources of uranium is about 5.4 million tonnes
 - That is the known amount of uranium
- The current usage in power reactors is about 68,000 tU/yr
- So there is about 80 years of uranium remaining in known sources



<http://www.world-nuclear.org/info/inf75.html>
 (updated December 2010)

Known Recoverable Resources of Uranium 2009

	tonnes U	percentage of world
Australia	1,673,000	31%
Kazakhstan	651,000	12%
Canada	485,000	9%
Russia	480,000	9%
South Africa	295,000	5%
Namibia	284,000	5%
Brazil	279,000	5%
Niger	272,000	5%
USA	207,000	4%
China	171,000	3%
Jordan	112,000	2%
Uzbekistan	111,000	2%
Ukraine	105,000	2%
India	80,000	1.5%
Mongolia	49,000	1%
Other	150,000	3%
World total	5,404,000	

Reasonably Assured Resources plus Inferred Resources, to US\$130/kg U
 from *Uranium 2009: Resources, Production and Demand*, OECD NEA & IAEA, 2009
 available from <http://www.oecd-nea.org/pub/uranium2009/>

U₃O₈ Prices



The Ux Consulting Company, LLC

(http://www.uxc.com/review/uxc_PriceChart.aspx?char=t=spot-u3o8-full)



The Ux Consulting Company, LLC

(http://www.uxc.com/review/uxc_Prices.aspx)

Principal Thorium-Containing Minerals

Mineral	Nominal composition	Examples of where found
Monazite	(La,Ce,Th)PO ₄	Brazil, India, Sri Lanka, Australia, South Africa, United States
Brockite	(Ca,Ce,Th)[(PO ₄)•(CO ₃)]•H ₂ O	United States
Thorianite	ThO ₂	Sri Lanka, Canada
Uranothorianite	(U,Th)O ₂	Malagasay Republic
Thorogummite	ThO ₂ •UO ₃	Brazil
Thorite	ThSiO ₄	Idaho and Montana
Auerlite	ThSiO ₄ •YPO ₄	Idaho and Montana
Uranothorite	(U,Th)SiO ₄	Blind River, Ontario
Brannerite	(U,Th,Ca ₂ ,Fe ₂)Ti ₂ O ₆	Blind River, Ontario
Bastnaesite	(La,Ce,Th)FCO ₃	California
Pyrochlore	(Na ₄ ,Ca ₂ ,U,Th)(Nb,Ta) ₄ O ₁₂	Colorado
Allanite	(Ca,Ce,Th) ₂ (Al,Fe,Mn,Mg) ₃ (SiO ₄) ₃ OH	Idaho and Montana

M. Benedict, T.H. Pigford and H.W. Levi, "Nuclear Chemical Engineering," McGraw-Hill Book Company (1981).

World Thorium Resources (economically extractable)

Country	Reserves (tonnes)
Australia	300,000
India	290,000
Norway	170,000
USA	160,000
Canada	100,000
South Africa	35,000
Brazil	16,000
Other countries	95,000
World total	1,200,000

US Geological Survey, *Mineral Commodity Summaries*, January 1999.



U₃O₈ (yellow cake) is the product after uranium mining and milling

Picture Courtesy: <http://biwii.blogspot.com>

Uranium Hexafluoride (UF₆) is white crystalline solid that resembles rock salt

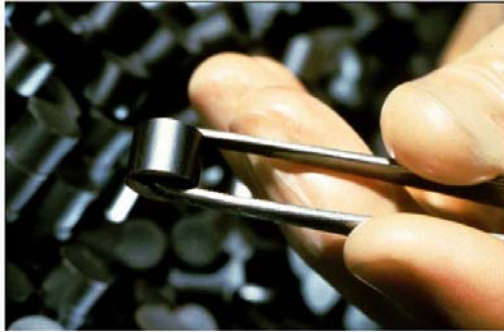
Picture Courtesy:
<http://web.ead.anl.gov/uranium/guide/prodhand/sld019.cfm>





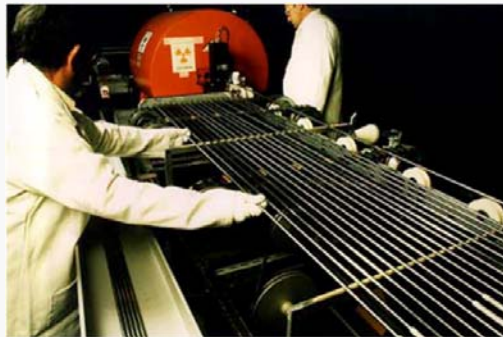
Uranium dioxide (UO₂) fuel pellets

Picture Courtesy: R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management, 2nd Edition*, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999



Uranium dioxide (UO₂) fuel pellets

Picture Courtesy: U.S. Nuclear Regulatory Commission (<http://www.nrc.gov/images/reading-rm/photo-gallery/20100907-005.jpg>)



UO₂ fuel pellets assembled into fuel rods

Picture Courtesy: U.S. Nuclear Regulatory Commission (<http://www.nrc.gov/images/reading-rm/photo-gallery/20100907-141.jpg>)

Fuel assemblies

Picture Courtesy: U.S. Nuclear Regulatory Commission (<http://www.nrc.gov/images/reading-rm/photo-gallery/20071114-045.jpg>)



Next Step in NFC: Fuel “Burning” (fission reaction) in Reactor Core



fuel assembly

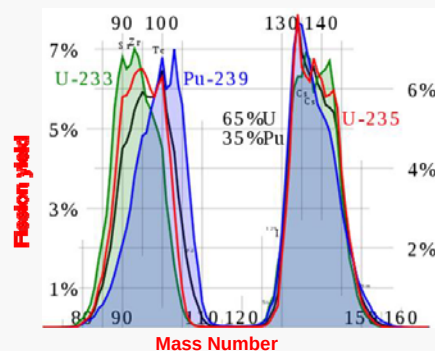


Picture of South Korea reactors (Yonggwang Site)

<http://www.icjt.org/npp/podrobnosti.php?drzava=16&lokacija=813>

Fission Product Buildup in Fuel

- A nuclide undergoing fission on average splits into two **fission products**
 - This also produces 2-3 neutrons and kinetic energy
- Fission products are typically **neutron rich** and most are highly **radioactive**
 - ^{137}Cs , ^{129}I , ^{90}Sr , ^{99}Tc , etc..
- You can touch fresh fuel
 - but spent fuel from reactor emits very harmful radiation and should be handled carefully
- Spent fuel largely consists of
 - ~96% depleted uranium
 - 1% plutonium
 - <1% other actinides
 - 3% fission products



Fission product yields by mass for thermal neutron fission of ^{235}U , ^{239}Pu

Picture Courtesy:
http://en.wikipedia.org/wiki/Fission_product_yield

Spent Fuel Storage (once through cycle)

- Discharged spent fuel is cooled for about 10-20 years in a large water pool (wet storage)
 - During wet storage fission products decay and radiation intensity reduces
- After 10-20 years cooling fuel assemblies are transferred into dry casks for long term storage (dry storage)



Onsite spent fuel pool (wet storage)
Picture courtesy: <http://world-nuclear.org>



Spent fuel dry storage in casks
Picture Courtesy: USNRC

Spent fuel assemblies being
moved in a spent fuel pool at Comanche
Peak

Photo courtesy: W. Charlton (2002).



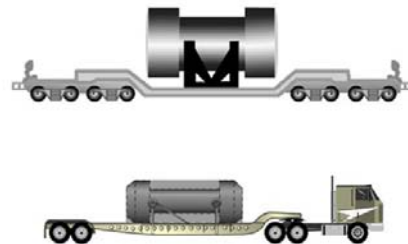


Spent Fuel Cask Transportation

(http://www.energy.gov/images/04774PD_casks-orange-vest.jpg)

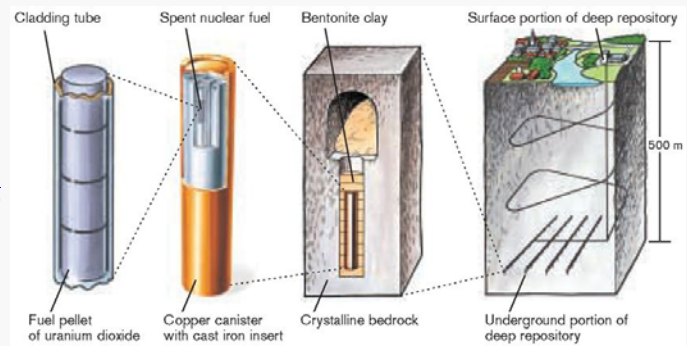
Comparison of Rail and Truck Spent Fuel Casks

(http://www.energy.gov/images/04774PD_truck_rail-car_a.jpg)



Deep Geological Repository in Sweden

(http://www.skb.se/upload/publications/pdf/Djupfor_eng.pdf)



Yucca Mountain Geological Repository Site in U.S.A.

(<http://www.nrc.gov/images/reading-rm/photo-gallery/20071115-018.jpg>)

Spent Fuel Storage (in closed fuel cycle)

- Spent fuel discharged from reactor is cooled for about 10 to 20 years in a large water pool (**wet storage**)
- Spent fuel undergoes **reprocessing** (chemical process) to separate fission products from depleted uranium and plutonium
- Separated uranium oxide and plutonium oxide is recycled as mixed oxide (**MOX**) fuel in reactors
- Separated fission products undergoes **high level waste management**

Spent fuel pool at La Hague
(<http://spectrum.ieee.org/energy/nuclear/nuclear-wasteland>)



HLW Storage at La Hague
(<http://www.aveva.com/EN/operations-1182/waste-management-at-aveva-la-hague.html>)

References

1. Nuclear Street (nuclear power portal)
<http://nuclearstreet.com>
2. World Nuclear Association, "Supply of Uranium,"
<http://www.world-nuclear.org/>
3. International Atomic Energy Agency, "Analysis of Uranium Supply to 2050," IAEA, Vienna, 2001
4. World Nuclear Association, "World Uranium Mining,"
<http://www.world-nuclear.org/>
5. The Ux Consulting Company: <http://www.uxc.com>

List of Required Texts

- The following textbook is required:
 - R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management, 2nd Edition*, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999
- This text is available for purchase online at <http://www.ans.org/store/vi-350015>
- Readings for the lectures and selected problem sets will be assigned from this text

Assignments, Activities, Exercises

- Reading Assignments
 - C&T Ch. 1, 2, 3
 - WNA papers: "Supply of Uranium," "World Uranium Mining," and "Uranium Enrichment"
 - IAEA paper "Uranium Forecast" and "In-Situ Mining Guide"
- Exercises
 - C&T Example 3.2 (Use Updated Values)
 - C&T Example 3.11 (Use Updated Values)
 - Develop the costs associated with the acquisition of nuclear fuel. Consider the costs associated with the acquisition of ore, mining, milling and conversion, as well as enrichment for a standardized PWR

End of Lecture 1

Uranium Mining, Milling, and Conversion Lecture 2

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Uranium Mining, Milling, and Conversion

- **Main Teaching Points**

- Uranium and Thorium mining methods
- Uranium and Thorium mining production
- Uranium milling and conversion process
- Uranium production and costs

Uranium Mining, Milling, and Conversion

- **Key Terms** – uranium exploration surveys, open pit mining, underground mining, in-situ leaching, uranium milling, yellow cake, U_3O_8 , uranium conversion, UF_6
- **Key Concepts** – uranium exploration methods, the process of nuclear fuel mining, how uranium ore is milled to U_3O_8 , the conversion process from U_3O_8 to UF_6 , the purpose of converting U_3O_8 to UF_6 , uranium costs

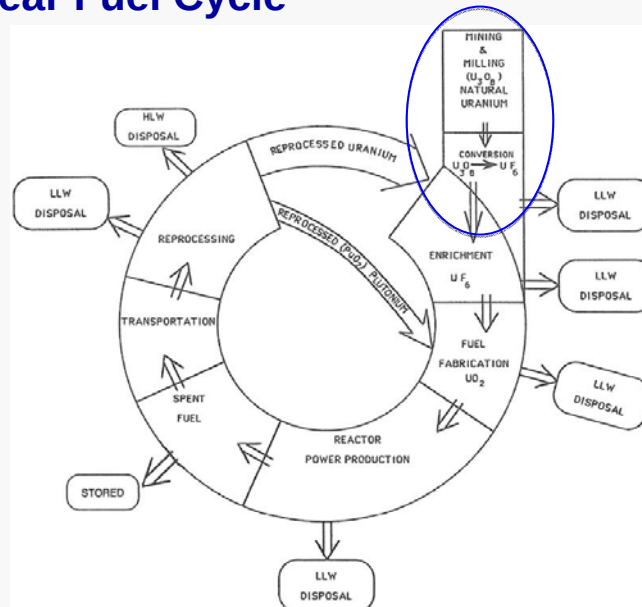
Uranium Mining, Milling, and Conversion

- **Desired Student Outcomes:**
 - Describe the major uranium mining methods
 - Explain the uranium milling and conversion processes
 - Summarize the scope and scale of uranium mining, milling, and conversion
 - Identify the costs of uranium

Assumptions and Prerequisites

- Successful completion of GNEII weeks one and two
- Bachelor's degree
- College level math

Nuclear Fuel Cycle



R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management, 2nd Edition*, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

Uranium Exploration to Find Economical Ores

- Geological studies
- Airborne surveys
- Surface surveys
- Hydro-chemical methods
- Well logging
- Botanical methods

Searching for uranium is in some ways easier than for other mineral resources because the radiation signature of uranium's decay products allows deposits to be identified and mapped from the air.

Uranium Mining Methods

- Open Pit
- Underground
- In-situ Leaching
- By-product

Open Pit Mining

- Careful planning and versatile equipment needed
- Drill holes at 50-ft centers may be needed to define the ore bed
- Topsoil is generally stockpiled for backfilling after mining
- After reaching the ore bed, the ore is dug with small front-end loaders and loaded into ore-hauling trucks to take to mill
- After removing the ore completely from the pit, the stockpiled topsoil is backfilled or the pit is left as it is for use as lakes



A truck carries uranium ore out of Ranger mine (Australia)
[Courtesy- Reuters: Rio Tinto/David Hancock, file photo]



Rossing Open pit mine
in Namibia

(http://en.wikipedia.org/wiki/File:Arandis_Mine_quer.jpg)

Underground Mining

- Similar to coal or any other mineral mining
- Opening to the mine may go down nearly a mile
- Side tunnels are cut into the ore deposits and ore is removed
- Ore is raised to the surface in mining carts and hauled by trucks to the mill
- There are several human hazards in underground mining, including radioactive radon gas



Photo Courtesy: Cotter Corporation.



Uranium Mine, Utah

(<http://en.wikipedia.org/wiki/File:UraniumMineUtah.JPG>)



DOSCO Roadheader

Photograph courtesy: Kennecott Uranium Company.

Electrically powered Continuous miner

Photograph courtesy: Kennecott Uranium Company.

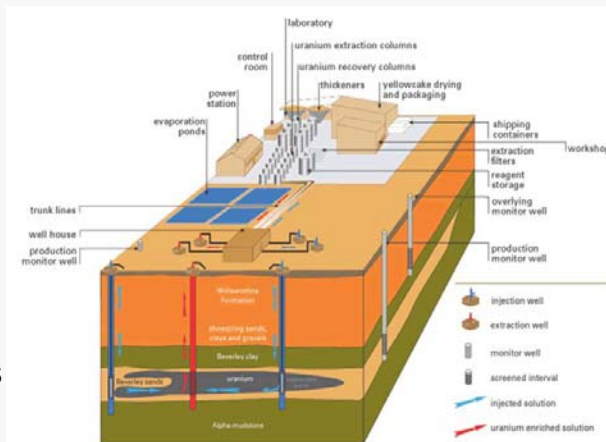


Underground uranium mining

(U.S. Environmental Protection Agency, "DOCUMERICA: The Environmental Protection Agency's Program to Photographically Document Subjects of Environmental Concern, compiled 1972 – 1977," 1972)

In-Situ Leaching (ISL)

- A leaching liquid is pumped into underground uranium deposits
- The uranium bearing liquid is pumped out
- ISL gains importance with a decrease in price of uranium
- Most U mining in US and Kazakhstan now done by ISL



ISL Process

(World Nuclear Association, <http://www.world-nuclear.org/info/inf27.html>)

ISL Pros and Cons



Advantages

- Eliminates
 - stockpiling and hauling of ore
 - crushing, grinding, and other milling operations
 - large-scale excavations
 - No need for large tailings piles
- Reduced risk to miners
- Very small fraction of radiation reaches the surface

Disadvantages

- Potential ground water contamination
- Low recovery rate (~50% or less)
- Unpredictable effects on host rock and restoration of natural conditions after leaching
- Production of some amounts of waste sludge and waste water



By-Product Mining

	Material	Location	Concentration Range (gU/MT)
Commercial Sources	Lignite	South Dakota, USA	10-4,000
	Shale	Sweden	250-325
	Gold tailings	South Africa	60-300
	Phosphate rock	Florida, USA	100-200
	Copper tailing	Western, USA	2-50
Sources not currently economical	Shale	Tennessee, USA	50-70
	Bostonite	Colorado, USA	33
	Granite	New Hampshire, USA	10-30
	Earth's crust	Average	1.7
	Seawater	Average	0.003

Thorium Mining

- Beach washings or dredge concentrate rich in mineral is concentrated using a host of spirals to enrich the feed with 97-98% heavy minerals
- Such upgraded material is dried in a fluid bed drier to take on the separation of individual minerals/ores by taking advantage of the difference in their electrical, magnetic properties as well as specific gravity
- Ilmenite, Garnet, Monazite (Thorium Mineral), Sillimanite, Rutile, Zircon are the typical minerals contained in beaches
- Minerals other than Monazite are of very good commercial value in chemical industry

Mining Methods

2005 World Production

open pit	30%
underground	38%
in situ leach (ISL)	21%
by-product	11%

2007 World Production

open pit	21%
underground	40%
in situ leach (ISL)	29%
by-product	10%

2006 World Production

open pit	24%
underground	41%
in situ leach (ISL)	26%
by-product	9%

2008 World Production

open pit & underground	62%
in situ leach (ISL)	28%
by-product	10%

2009 World Production

open pit & underground	57%
in situ leach (ISL)	36%
by-product	7%

Source: <http://www.world-nuclear.org/info/inf23.html>

In 2009, ten companies marketed 89% of the world's uranium mine production:

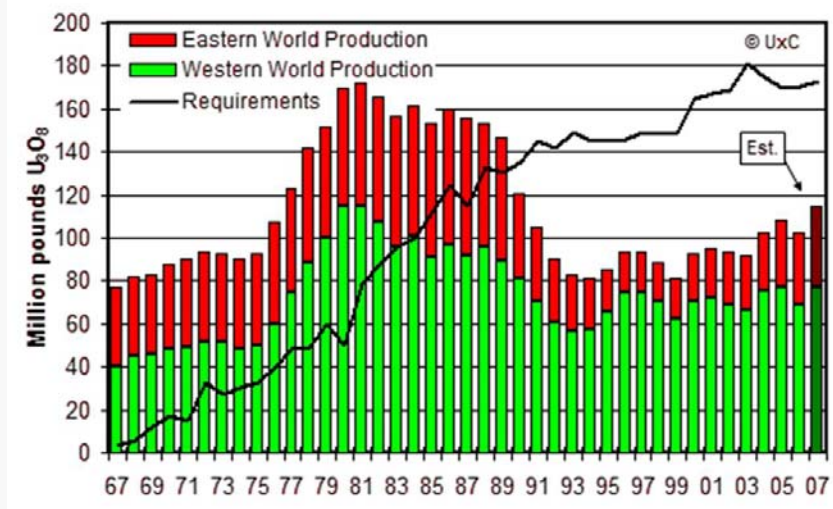
Company	tonnes U	%
Areva	8623	17
Cameco	8000	16
Rio Tinto	7963	16
KazAtomProm	7467	15
ARMZ	4624	9
BHP Billiton	2955	6
Navoi	2429	5
Uranium One	1368	3
Paladin	1210	2
GA/Heathgate	583	1
Other	5550	11
Total	50,772	100%

Source: <http://www.world-nuclear.org/info/inf23.html>

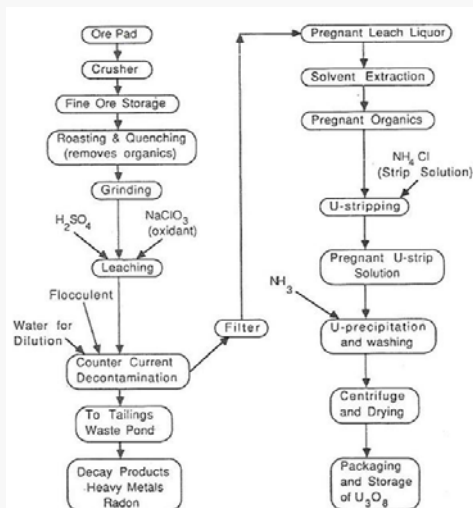
U Production from Mines (tonnes U)

Country	2003	2004	2005	2006	2007	2008	2009
Kazakhstan	3300	3719	4357	5279	6637	8521	14020
Canada	10457	11597	11628	9862	9476	9000	10173
Australia	7572	8982	9516	7593	8611	8430	7982
Namibia	2036	3038	3147	3077	2879	4366	4626
Russia	3150	3200	3431	3400	3413	3521	3564
Niger	3143	3282	3093	3434	3135	3032	3243
Uzbekistan	1589	2016	2300	2270	2320	2338	2429
USA	779	878	1039	1692	1654	1430	1453
Ukraine	800	800	800	800	846	800	840
China	750	750	750	750	712	769	750
South Africa	758	755	674	534	539	655	563
Brazil	310	300	110	190	299	330	345
India	230	230	230	230	270	271	290
Czech Rep	452	412	408	359	306	263	258
Other	248	219	236	200	167	127	237
Total	35576	40178	41719	39670	41282	43853	50772
Percentage of World Demand		65%	63%	64%	68%	76%	

Source: <http://www.world-nuclear.org/info/inf23.html>



Uranium Milling



Uranium Milling Process Flow Chart:

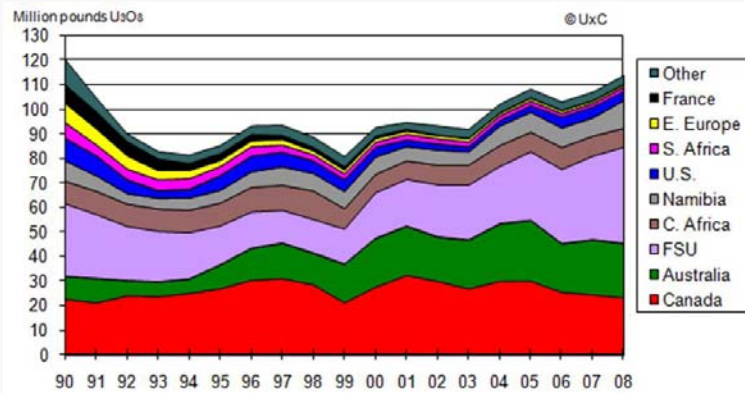
R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management*, 2nd Edition, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999



U_3O_8 (yellow cake) is the product after uranium mining and milling
Strictly speaking yellow cake is a mixture concentrated in U_3O_8 and pure U_3O_8 is dark gray in color

Picture Courtesy: <http://biwii.blogspot.com>

Historical U_3O_8 production charts

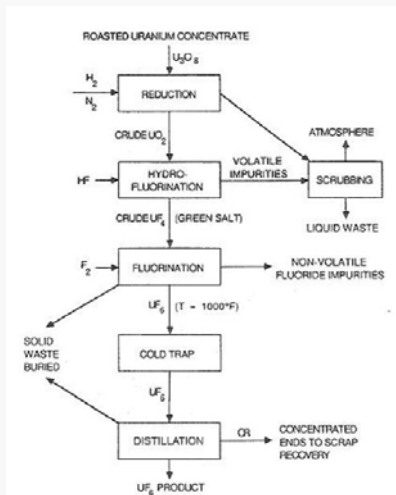


Source: The Ux Consulting Company, LLC and include a link back to UxC's home page at <http://www.uxc.com/>

Fuel Quantity & Cost Calculator
http://www.uxc.com/tools/uxc_FuelCalculator.aspx

Uranium Conversion

The Dry Hydrofluor Process Flow Chart to Convert U_3O_8 to UF_6



UF₄ (Green Salt):

<http://lewis.armscontrolwonk.com/images/623.gif>

Yellow cake by reduction with hydrogen gives UO_2 , which is converted with hydrofluoric acid (HF) to uranium tetrafluoride (UF_4)

Oxidation with fluorine yields UF_6

Uranium Conversion Process Flow Chart
 (R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management*, 2nd Edition, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999)

World Conversion Nameplate Capacity

Company (Country)	Nameplate Capacity as of Dec 31, 2004 (Million kgU)
BNFL (UK) 	6.0
Cameco (Canada) 	12.5*
CNNC (China) 	1.0
Comurhex (France) 	14.0
ConverDyn (USA) 	14.0
IPEN (Brazil) 	0.1
Minatom (Russia) 	22.0
Total	69.6
* Additional Cameco capacity includes 2,800 tonnes per year of UO ₂ and 2,000 tonnes as metal	

References

1. International Atomic Energy Agency, "Manual of Acid in Situ Leach Uranium Mine Technology," IAEA-TECDOC-1239, Vienna, 283pp, 2001
2. World Nuclear Association, "Supply of Uranium," <http://www.world-nuclear.org/>
3. International Atomic Energy Agency, "Analysis of Uranium Supply to 2050," IAEA, Vienna, 2001
4. World Nuclear Association, "World Uranium Mining," <http://www.world-nuclear.org/>

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 - R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management, 2nd Edition*, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999
- This text is available for purchase online at <http://www.ans.org/store/vi-350015>
- Readings for the lectures and selected problem sets will be assigned from this text

Assignments, Activities, Exercises

- Reading Assignments
 - C&T Ch. 1, 2, 3
 - WNA papers: "Supply of Uranium," "World Uranium Mining," and "Uranium Enrichment"
 - IAEA paper "Uranium Forecast" and "In-Situ Mining Guide"
- Exercises
 - C&T Example 3.2 (Use Updated Values)
 - C&T Example 3.11 (Use Updated Values)
 - Develop the costs associated with the acquisition of nuclear fuel. Consider the costs associated with the acquisition of ore, mining, milling and conversion, as well as enrichment for a standardized PWR

End of Lecture 2

Uranium Enrichment Lecture 3

GNEII Pilot Course

February – May 2011

Khalifa University of Science, Technology and Research,

Abu Dhabi, United Arab Emirates

(William Charlton)

Texas A&M University, USA

SAND 2010_____

Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

Uranium Enrichment

- **Main Teaching Points:**

- What is isotope enrichment?
- Types of enrichment technologies
- Separative work unit
- Energy consumption difference between enrichment processes

Uranium Enrichment

- **Key Terms** – isotope separator, feed product and waste, assay, enrichment element, enrichment stage, enrichment cascades, diffusion/diffusors, centrifuge, AVLIS, MLIS, EMIS, Calutron, gaseous diffusion enrichment, gas centrifuge enrichment, separative work unit (SWU)
- **Key Concepts** – the two common enrichment methods (gaseous diffusion and the use of centrifuges), small scale methods (AVLIS & MLIS), impact of enrichment on uranium costs

Uranium Enrichment

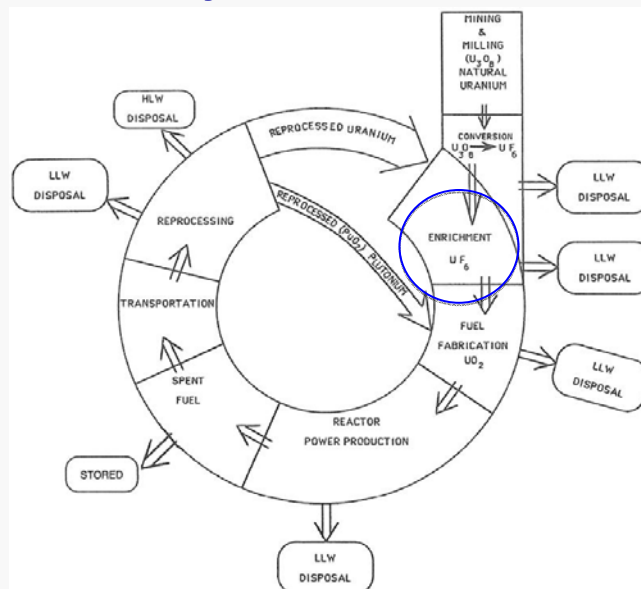
- **Desired Student Outcomes:**

- Briefly describe various ^{235}U enrichment processes
- Explain the advantages of using gaseous centrifuges compared to diffusers
- Describe the enrichment process in terms of feed, product, and waste
- Recognize the engineering challenges associated with large-scale uranium enrichment
- Explain separative work unit (SWU) and its impact on uranium costs

Assumptions and Prerequisites

- Successful completion of GNEII weeks one and two
- Bachelor's degree
- College level math

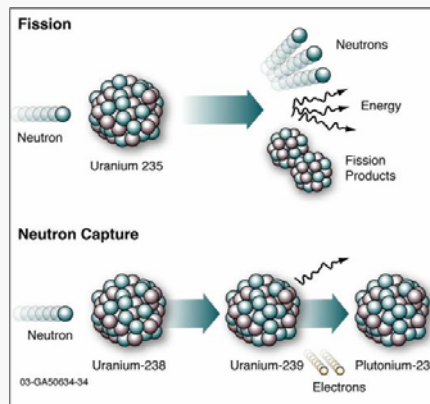
Nuclear Fuel Cycle



R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management*, 2nd Edition, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

Uranium Basics

- Natural uranium has three isotopes:
 - 99.2745% ^{238}U
 - 0.72% ^{235}U
 - 0.0055% ^{234}U
- ^{235}U is the key content in the uranium fuel needed to produce energy and in turn electricity
 - enriching ^{235}U content in nuclear fuel from 0.72% to about ~5% is needed to operate light water reactors
- ^{238}U absorbs a neutron while in reactor and eventually turns into ^{239}Pu
 - which is fissile but not found in nature



Fission and Radiative Capture Reactions

(<http://www.criticalissuesforum.org/FILES/WKSHP2007/essner.ppt>)

Enrichment

- Several technologies have been developed to perform isotopic enrichment
 - Diffusion
 - Centrifuge
 - Laser – AVLIS, MLIS
 - Other – aerodynamic, electromagnetic

Table 1.2 Uranium enrichment techniques, according to physical principles and mechanisms used

Property of isotope	Physical principle	Physical mechanism	Separation process
Nuclear mass	Newton's law of acceleration	Acceleration by electro-magnetic forces	Calutron Plasma centrifuge Ion cyclotron resonance
		Shift in chemical equilibrium	Chemical exchange
	Equipartition of energy	Selective absorption of infra-red light	Molecular laser isotope separation (MLIS)
		Flow through porous barrier	Gaseous diffusion
Nuclear size, shape, spin, magnetism	Quantum theory of atomic structure	Diffusion against pressure gradient	Centrifuge Jet nozzle Vortex tube
		Selective absorption of visible or ultraviolet light	Atomic vapour laser isotope separation (AVLIS)

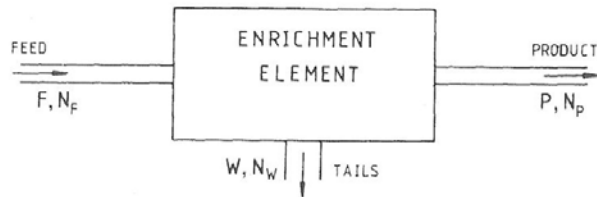
A. Krass et al., *Uranium Enrichment and Nuclear Weapons Proliferation*, Taylor & Francis Ltd., London, Pg 8, 1983.

Basic Concept of an Enrichment Element

- To characterize the efficiency of an isotope enrichment process a term called **separation factor** is used
 - Higher the value of the separation factor, the easier to enrich

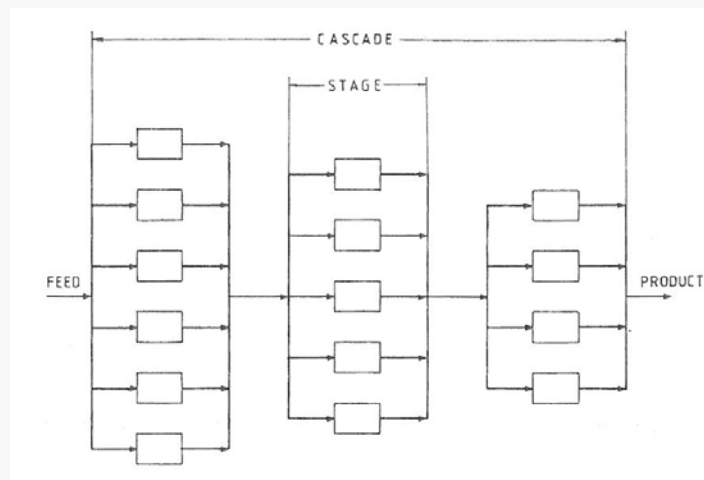
Figure 5.1. An enrichment element

Schematic of an enrichment element showing the input (feed= F) and outputs (product= P , tails= W). N_F , N_P , N_W denote the fraction of ^{235}U present in each stream.



A. Krass et al., *Uranium Enrichment and Nuclear Weapons Proliferation*, Taylor & Francis Ltd., London, pg. 95, 1983.

Stages and Cascades



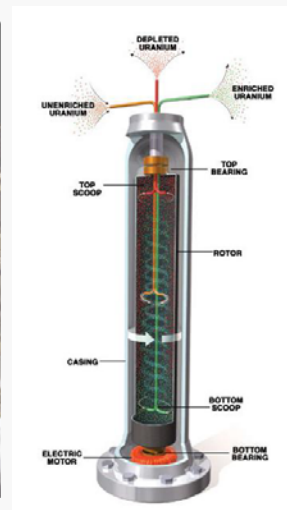
Separative Work Unit (SWU)

- The amount of physical effort needed to reach a certain enrichment is expressed in terms of Separative Work Units (SWU)
- The amount of SWU expended is related to the increased cost of the uranium
- Fuel Quantity & Cost Calculator:
 - http://www.uxc.com/tools/uxc_FuelCalculator.aspx

Centrifuge Enrichment

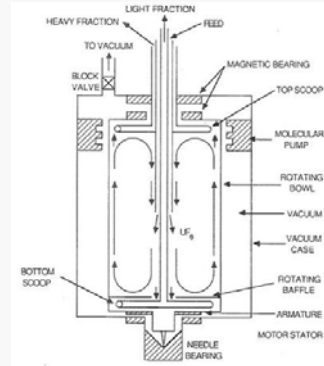


<http://www.criticalissuesforum.org/FILES/WKSHP2007/essner.ppt>



Centrifuge Enrichment Technique

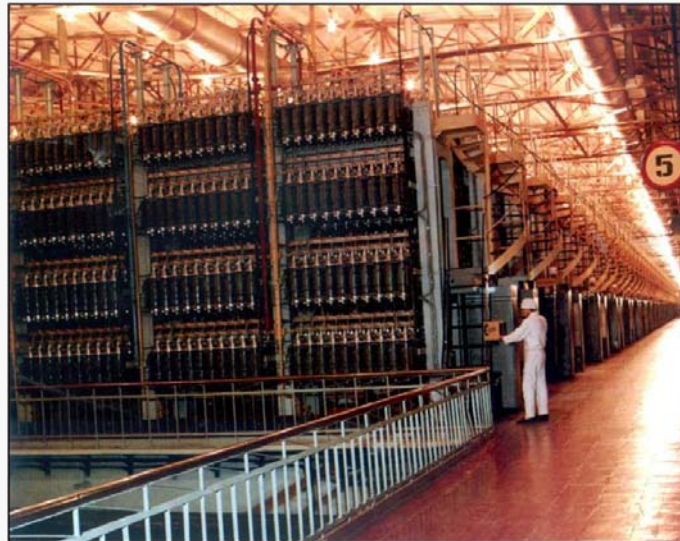
- The UF_6 gas rotating inside the cylinder is subject to centrifugal acceleration thousands of times greater than gravity
 - due to this, pressure at the outer radius of the cylinder will be million times greater than at the center
 - results in appreciably higher relative abundance of the heavier isotope at the outer radius than at the center
- A countercurrent flow induced between the ^{235}U -depleted stream (near the outer radius) and the ^{235}U -enriched stream (near the axis)
 - enables the composition difference between top and bottom of the cylinder
- The energy per SWU for a gas centrifuge is about 1/10 that of a diffusion process



Countercurrent gas centrifuge with internal circulation

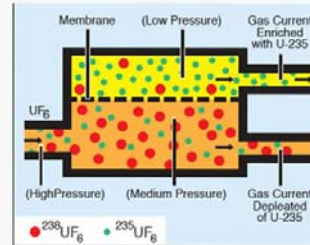
(R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management, 2nd Edition*, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999)

Small Russian Gas Centrifuges



Gaseous Diffusion

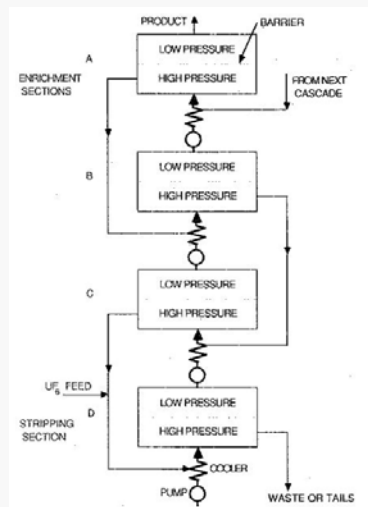
- When UF_6 gas effuse through tiny pores in a thin barrier subjected to a pressure difference,
 - the effusion of lighter molecules ($^{235}\text{UF}_6$) is faster than that of heavy molecules ($^{238}\text{UF}_6$)
 - and the gas passing through the barrier is slightly enriched in $^{235}\text{UF}_6$
- The separation factor for diffusion process is 1.003
- Energy consumption for gaseous diffusion is about 2.3–3.0 MW-hr per SWU Produced



Basic Diffusion Process

(<http://www.chemcases.com/nuclear/nc-07.html>)

Steps in Gaseous Diffusion Enrichment



Typical diffusion cascade

(R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management*, 2nd Edition, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999)

Uranium Enrichment using Atomic Vapor Laser Isotope Separation (AVLIS) Process

- ^{235}U atoms and ^{238}U atoms absorb light (LASER) of different frequencies (or colors)
- The right amount of absorbed energy can ionize one isotope and not the other
- Ionized uranium isotope gas atoms can be collected on negatively charged plate

Separation factor for AVLIS is ~ 7 and complete separation is obtained in one stage
Low energy consumption of the order of 0.3 MWh/SWU

Molecular Laser Isotope Separation (MLIS) Process

- Preferential absorption of LASER energy by $^{235}\text{UF}_6$ can be dissociate into F and $^{235}\text{UF}_5$
- $^{235}\text{UF}_5$ thus formed precipitates to a solid powder that can be filtered from the gas stream

References

1. International Atomic Energy Agency, "Manual of Acid in Situ Leach Uranium Mine Technology," IAEA-TECDOC-1239, Vienna, 283pp, 2001
2. World Nuclear Association, "Supply of Uranium," <http://www.world-nuclear.org/>
3. International Atomic Energy Agency, "Analysis of Uranium Supply to 2050," IAEA, Vienna, 2001
4. World Nuclear Association, "World Uranium Mining," <http://www.world-nuclear.org/>
5. World Nuclear Association, "Uranium Enrichment," <http://www.world-nuclear.org/>

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End of Lecture 3

Week 3 - Day 2

Nuclear Fuel Cycle

Lecture 4 – Nuclear Fuel Design and Fabrication

Lecture 5 – Power Operations - 1

Lecture 6 – Power Operations - 2

Nuclear Fuel Design and Fabrication Lecture 4

GNEII Pilot Course

February – May 2011

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Nuclear Fuel Design and Fabrication

- **Main Teaching Points**

1. Nuclear fuel design
2. Fuel fabrication starting from enriched UF_6
3. Characteristics of fuel and cladding
4. Fuel behavior during operation and its effect on the fuel design
5. Sample fuel assemblies

Nuclear Fuel Design and Fabrication

- **Key Terms** – nuclear fuel, fuel pellet, fuel fabrication, thermal reactor, fast reactor, heavy water reactor, light water reactor, neutron absorbers
- **Key Concepts** – constituents of nuclear fuel assembly, steps of nuclear fuel fabrication, importance of nuclear fuel design, importance of fuel cladding, desirable characteristics of nuclear fuel material, purpose of burnable neutron absorber rods

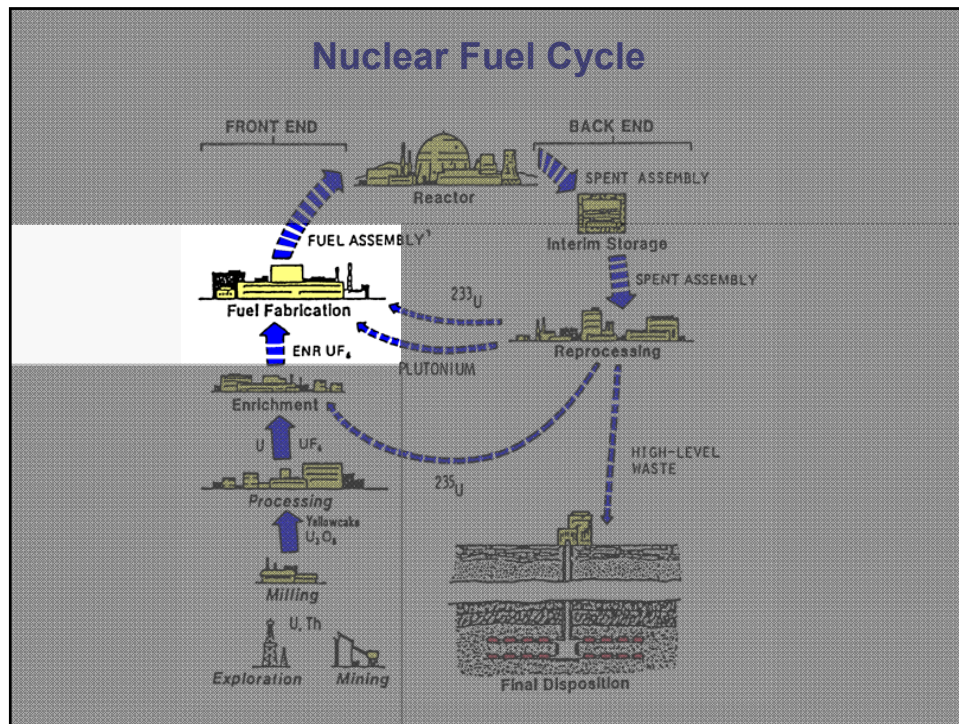
Nuclear Fuel Design and Fabrication

- **Desired Student Outcomes:**

1. Importance of nuclear fuel design
2. Study about reactor types and respective fuel types
3. Be familiar with the major steps and intermediate products associated with power reactor fuel fabrication
4. Know the importance of the fuel rod cladding and gas gap
5. Understand the constituents of nuclear fuel assembly
6. Reason for neutron absorber rods use in some of the fuel assemblies

Assumptions and Prerequisites

- Successful completion of GNEII weeks one and two
- Bachelor's degree
- College level math



Why nuclear fuel performance is important?

- Construction of nuclear power plants is costly (**high capital cost**)
- Should be compensated by **low running cost** to compete with other electricity generation methods
- Plant operations should be reliable and uninterrupted for long duration
- **One important aspect on which uninterrupted long term nuclear plant operation depends is its fuel design and fabrication**
- Fuel failures may cause plant shutdowns

What constitutes a nuclear fuel?

- Contain at least one “**fissile material**” either uranium or plutonium isotopes (^{235}U , ^{233}U , & ^{239}Pu , which produces energy)
- Usually contain a “**fertile material**” either uranium or thorium (^{238}U , ^{232}Th)
- A small fraction of fertile material is converted to fissile material while inside the reactor ($^{238}\text{U} \rightarrow ^{239}\text{Pu}$; $^{232}\text{Th} \rightarrow ^{233}\text{U}$)
- Fuel could be ceramic (oxide, carbide, nitride), metallic, etc. **Currently used nuclear fuel form in most reactors is uranium dioxide (UO_2) ceramic**
- Natural UO_2 has both fissile (^{235}U) and fertile (^{238}U) content
- Enrichment increases the fissile (^{235}U) content and reduces the fertile (^{238}U) content. **Typical ^{235}U enrichments of ~3% to 5% are used in light water reactor (LWR) fuel**
- There are a handful of nuclear power reactors using **mixed oxide (MOX) fuel** – [mixture of UO_2 - PuO_2]

Types of nuclear power plants (NPPs) & their fuel

- In general NPPs are classified into types
 - **Thermal reactors**
 - **Fast reactors**
- Classification is based on the **speed of neutrons**, which produces most number of nuclear fissions in a reactor
- Most neutrons in a **thermal** reactor has a speed of **2.2 km/s**
- Most neutrons in a **fast** reactor has a speed in the range from **3500 km/s to 7000 km/s**
- Two major kinds of thermal reactors
 - **heavy water reactors**
 - (use natural UO_2 fuel)
 - Canadian CANDU reactor is an example
 - **light water reactors (LWRs)**
 - use enriched UO_2 fuel - ^{235}U enrichments of ~3% to 5%
 - LWRs are classified further into boiling water reactors (BWRs) and pressurized water reactors (PWRs)
- Fast reactors usually use MOX fuel [mixture of UO_2 - PuO_2]

Nuclear Fuel Cladding & Assembly

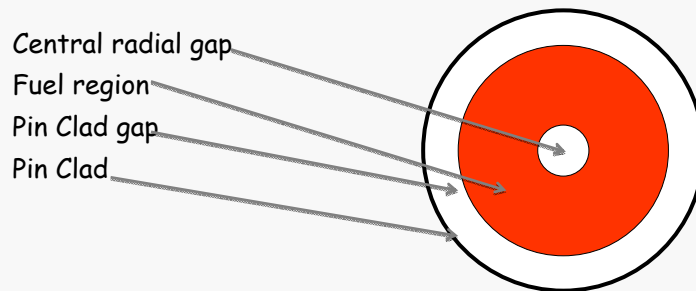
- Fuel pellets are held together inside a fuel pin tube (cladding) usually made out of metal alloys
 - LWRs usually use Zircaloy as cladding material
 - Fast reactors use stainless steel alloy as cladding material
- Fuel cladding
 - Prevents corrosion of the fuel by the reactor coolant
 - Retains harmful fission products
 - Plenum inside the fuel rod allows fission gas (Xe, Kr, He) retention
 - Provides surface for heat transfer from fuel to coolant
 - Accommodates fuel expansion
- Fuel cladding properties
 - Corrosion resistant
 - Very small neutron absorption
 - Good heat transfer characteristics



Picture Courtesy: <http://nrc.gov>

How much uranium fuel is needed?

- A typical example of 1000 MWe PWR
 - ~ 70 tons of uranium in core in about ~ 160 fuel assemblies
 - 18 months fuel cycle
 - ~ 50 fuel assemblies refueled per equilibrium fuel cycle
 - ~ 25 tons of uranium per 18 months (equilibrium fuel cycle)
- A 1000 MWe coal fueled power station needs **a few million tons of coal** per year



Desirable Nuclear Fuel Characteristics

- High melting point
- High thermal conductivity (heat transfer property)
- Chemical and structural stability
- No reaction with high-temperature water (reactor coolant)
- Fission products retention capability
- Low fabrication cost

Note: UO_2 fuel characteristics

- Melting point 2865°C
- Theoretical density 10.97 g/cm^3
- Thermal conductivity
 - $4.777 \times 10^{-3} \text{ W/m.K}$ at 20°C
 - $1.91 \times 10^{-3} \text{ W/m.K}$ at 1000°C

Zircaloy cladding characteristics

- Melting point 1845°C
- Tin+iron+chromium+nickel $<2\%$
- Thermal conductivity
 - $\sim 13 \text{ W/m.K}$ at 20°C
 - $\sim 27 \text{ W/m.K}$ at 1000°C

Tables Courtesy:

R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management*, 2nd Edition, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

UO_2 Fuel Fabrication Steps (1 of 2)

1. Heat cylinder of UF_6 ($\sim 100^\circ\text{C}$) and the vapor is transported using a carrier gas (dry nitrogen/argon)
2. Bubble UF_6 gas through water $\rightarrow \text{UO}_2\text{F}_2$ in water (exothermic reaction, you need to remove heat with a heat exchanger)
3. Add NH_4OH to water to precipitate Ammonium Di-Uranate (ADU), $(\text{NH}_4)_2\text{U}_2\text{O}_7$ ($\sim 60^\circ\text{C}$)
4. Dry ADU at ~ 250 degree Celsius (Calcination $\sim 700^\circ\text{C}$) $\rightarrow \text{UO}_3$; reduce further with hydrogen $\rightarrow \text{UO}_2$ $\sim 625^\circ\text{C}$ and further stabilize UO_2 in controlled air atmosphere for it not to oxidize back to UO_3

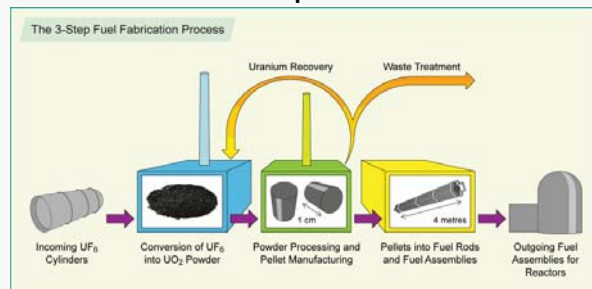


Image Courtesy:

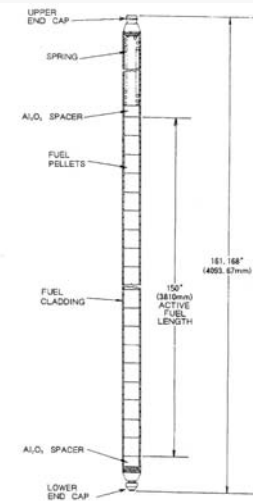
http://www.world-nuclear.org/info/nuclear_fuel_fabrication-inf127.html

UO₂ Fuel Fabrication Steps (2 of 2)

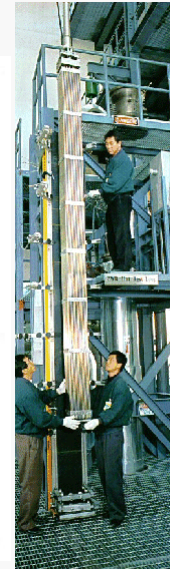
5. Grind UO₂ to fine, uniform powder (under inert atmosphere to prevent oxidization)
6. Add adhesive and press into cylindrical pellet, "green pellet"
7. Sinter green pellets at 1700° C in hydrogen atmosphere → densification
8. Load into cladding (fuel rods)
9. Assemble fuel bundle

Fuel Rod Image Courtesy:

http://www.kntc.re.kr/openlec/nuc/NPRT/module2/module2_2/module2_2_2/images/p45.jpg



Schematic of PWR fuel rod



PWR fuel assembly inspection

Image Courtesy:

<http://www.nucleartourist.com/systems/pwrfuel1.htm>

Powerful Pellet

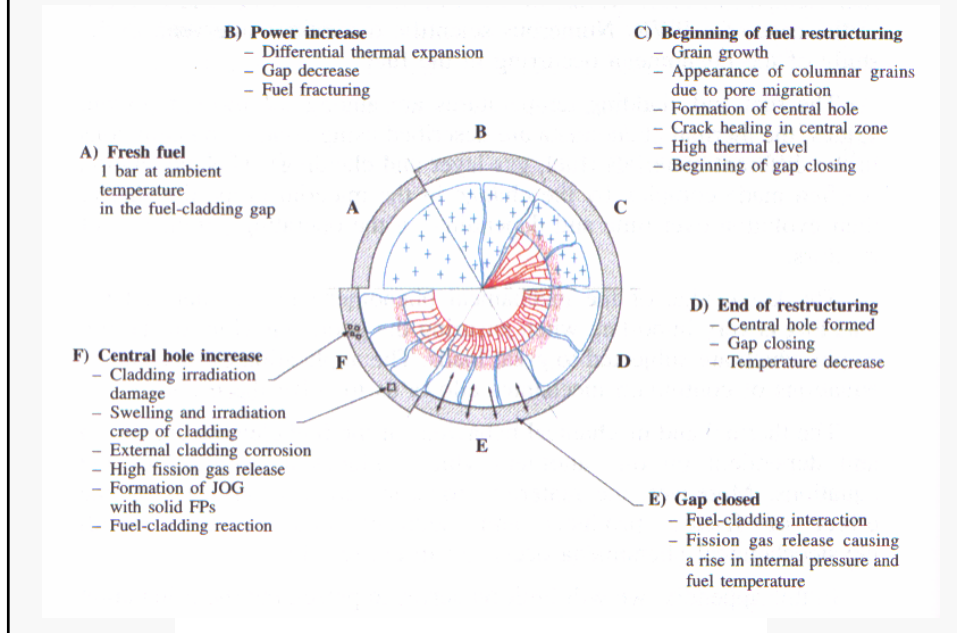
A typical pellet of uranium weighs about 7 grams (0.24 ounces). It contains as much energy as...

3.5 barrels of oil or...	17,000 cubic feet of natural gas, or...	1,780 pounds of coal.

Source: Nuclear Energy Institute

To avoid swelling of the fuel the **sintered density** of UO₂ pellets is kept in the range from 90% to 95% of **theoretical density** (10.97 g/cm³)

Fuel Pellet Life Cycle



Neutron Absorbers in Fuel Assembly

- Neutron absorber rods are necessary to be incorporated in the fuel to accomplish efficient reactor control
- A certain quantity of such neutron absorbers is placed in the core
- As the reactor operates, the amount of the absorber is reduced, i.e. burned as it absorbs neutrons

Purpose of neutron absorbers

- To control fission reactions in the excess fissile material loaded in the core to compensate for fuel depletion
- To compensate for the negative effect on fission product neutron absorbers and fuel temperature changes
- Compounds of boron (B_4C), gadolinium (Gd_2O_3) and erbium (Er_2O_3) are usually used in neutron absorber rods

Example Fuel Assemblies - 1

PWR Fuel Assembly Schematic

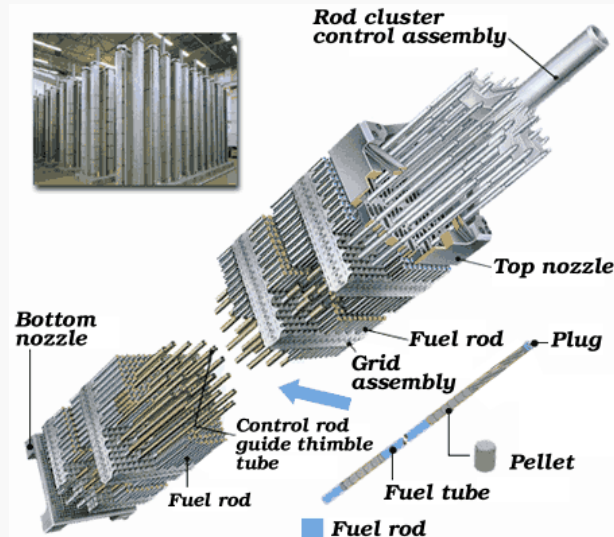


Image Courtesy: <http://www.mnf.co.jp/pages2/pwr2.htm>

Example Fuel Assemblies - 2

VVER 1000 (Russian PWR) Fuel Assembly (~435 kg U) Data & Picture

Parameter	Value
Fuel pins	
Number of fuel pins	312
Number of guide tubes	18
Number of instrumentation tubes	1
Pin pitch, cm	1.275
Fuel rods	
Pellet diameter, cm	0.772
Clad inside diameter, cm	0.772
Clad outside diameter, cm	0.910
Clad material	Zr
Active fuel length, cm	353.0
Guide tubes	
Inside diameter, cm	1.090
Outside diameter, cm	1.265
Material	Zr
Central instrumentation tube	
Inside diameter, cm	0.960
Outside diameter, cm	1.125
Material	Zr



Image Courtesy:
<http://www.elemash.ru/comm/on/img/uploaded/5.jpg>



Two CANDU reactor fuel bundles: Each about 50 cm in length and 10 cm in diameter, and generating about 1 GWh of electricity during its time in the reactor. Photo courtesy of Atomic Energy Canada Limited

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References

1. International Atomic Energy Agency, "Manual of Acid in Situ Leach Uranium Mine Technology," IAEA-TECDOC-1239, Vienna, 283pp, 2001
2. International Atomic Energy Agency, "Report on the preliminary fact-finding mission following the accident at the nuclear fuel processing facility in Tokaimura, Japan," 1999
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8. W. Charlton and D. Boyle, Course Presentation Slides, Private Communication

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- Readings for the lectures and selected problem sets will be assigned from this text

Assignments, Activities, Exercises

- Reading Assignments
 - C&T Ch. 4, 6
 - Assorted papers
- Exercises
 - Ragheb Ch. 10 problems 1-4
- Activity
 - Why not lump uranium in the core instead of having pins and assemblies?

End of Lecture 4

Power Operations - 1

Lecture 5

GNEII Pilot Course

February – May 2011

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Abu Dhabi, United Arab Emirates

William Charlton

Texas A&M University, USA

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Power Operations - 1

- **Main Teaching Points**

1. Nuclear Reactor Generations
2. Components of a Nuclear Reactor
3. Key Technical Terms used in Nuclear Power Operation
4. Technical Information about typical PWR, BWR, HWR and Fast Reactors

Power Operations - 1

- **Key Terms** – nuclear reactor generations, fuel, moderator, coolant, steam generator, control rods, pressure vessel, containment, neutron multiplication factor, reactivity, rated power, thermal power, gross power, net power, burnup, PWR, BWR, CANDU(HWR), FBR
- **Key Concepts** –key technical terms used to characterize reactor power operation, components of a nuclear reactors, technical data on typical operating nuclear power plants

Power Operations - 1

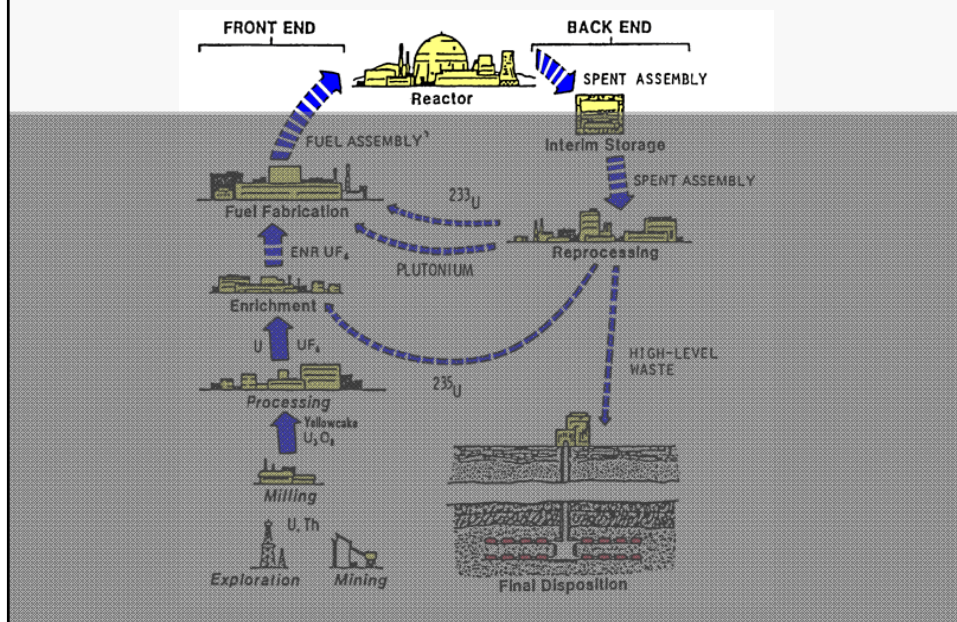
- **Desired Student Outcomes:**

1. Understand the key components of a reactor
2. Knowledge about the technical terms used to describe the operational characteristics of reactor
3. Understand the differences between PWR, HWR and Fast Reactors

Assumptions and Prerequisites

- Successful completion of GNEII weeks one and two
- Bachelor's degree
- College level math

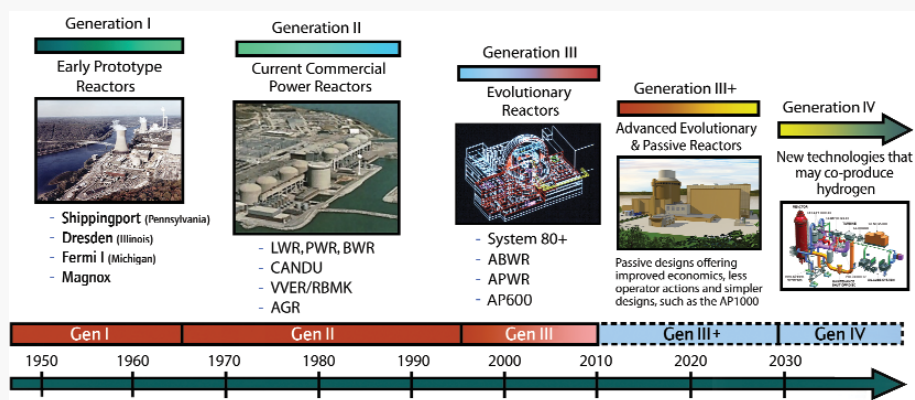
Nuclear Fuel Cycle



Introduction

- **Electric power generation** in nuclear power plants (NPPs) is facilitated by utilizing **fission heat energy** generated in the fuel
- Fission heat generated in the fuel is transferred to **coolant**, which in turn produces **steam** with the help of heat exchangers
- High pressure steam produced will be used to **rotate the turbine** to produce electricity
- Nuclear fuel used is dependent upon the reactor type and operating cycle length
 - Power
 - PWR, BWR, HWR, FBR
- **Sophisticated calculations** are performed to analyze reactor core physics for arriving at the **amount of fuel to be used**, **irradiation time**, **operating specifications**, **in-core fuel management**
- Nuclear power operation should address safety requirements and economics

Nuclear Reactor Generations



Components of a Nuclear Reactor

Fuel. Usually pellets of UO_2 arranged in tubes to form fuel rods. The rods are arranged into fuel assemblies in the reactor core.

Moderator. This is material in the core which slows down the fast neutrons released from fission so that they cause more fission. It is usually water, but may be heavy water or graphite.

Control rods. These are made with neutron-absorbing material such as cadmium, hafnium or boron, and are inserted or withdrawn from the core to control the rate of reaction, or to halt it.

Coolant. A liquid or gas circulating through the core so as to transfer the fission heat from it. In LWRs, water moderator functions also as primary coolant.

Pressure vessel or pressure tubes. Usually a robust steel vessel containing the reactor core and moderator/coolant, but it may be a series of tubes holding the fuel and conveying the coolant through the moderator.

Steam generator. (not in BWR) Part of the cooling system where the primary coolant bringing heat from the reactor is used to make steam for the turbine.

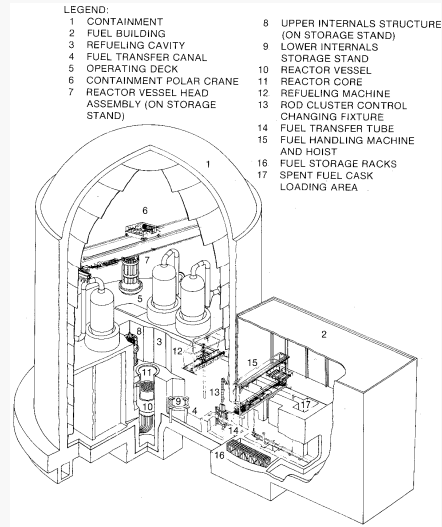
Containment. The structure around the reactor core which is designed to protect it from outside intrusion and to protect those outside from the effects of radiation in case of any malfunction inside. It is typically a meter-thick concrete and steel structure.

Nuclear power plants in commercial operation

Reactor type	Main Countries	Number	GWe	Fuel	Coolant	Moderator
Pressurised Water Reactor (PWR)	US, France, Japan, Russia, China	265	251.6	enriched UO_2	water	water
Boiling Water Reactor (BWR)	US, Japan, Sweden	94	86.4	enriched UO_2	water	water
Pressurised Heavy Water Reactor 'CANDU' (PHWR)	Canada	44	24.3	natural UO_2	heavy water	heavy water
Gas-cooled Reactor (AGR & Magnox)	UK	18	10.8	natural U (metal), enriched UO_2	CO_2	graphite
Light Water Graphite Reactor (RBMK)	Russia	12	12.3	enriched UO_2	water	graphite
Fast Neutron Reactor (FBR)	Japan, Russia	2	1.0	PuO_2 and UO_2	liquid sodium	none
Other	Russia	4	0.05	enriched UO_2	water	graphite
	TOTAL	439	386.5			

Source: <http://www.world-nuclear.org/info/inf32.html>

Cutaway View of PWR



Source: "The Westinghouse Pressurized Water Reactor Nuclear Power Plant," Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, pg. 184 (1984).

Key Terms

NPP power outputs are quoted in three ways:

Thermal Power (MWt), which depends on the design of the actual nuclear reactor itself, and relates to the quantity and quality of the steam it produces.

Gross Electrical Power (MWe) indicates the power produced by the attached steam turbine and generator, and also takes into account the ambient temperature for the condenser circuit (cooler means more electric power, warmer means less). **Rated gross power** assumes certain conditions with both.

Net Electrical Power (MWe), which is the power available to be sent out from the plant to the grid, after deducting the electrical power needed to run the reactor (cooling and feed-water pumps, etc.) and the rest of the plant.*

Key Terms

- Rated Power (P_0)
- Burnup (energy produced per unit mass)

$$BU\left(\frac{MWd}{MTU}\right) = \frac{P_0(MWth) \cdot CF \cdot T(d)}{MTU}$$

- Capacity Factor (fraction of rated power produced)

$$CF = \int_0^T \frac{P(t)}{P_0 T} dt$$

Key Terms

- Availability Factor

$$AF = \frac{\text{Time during which plant was operational in period } T}{T}$$

- Effective Full Power Days (EFPD)

$$EFPD = \int_0^T CF(t) dt$$

- Neutron flux (a quantification for neutron population in reactor) expressed as neutrons/cm²/sec [n.cm².s⁻¹]

Key Terms

Neutron multiplication factor, k

- One fission reaction by a neutron with a ^{235}U nucleus in fuel can produce about 2 to 3 neutrons
- These 2 to 3 neutrons could be absorbed
 - by a ^{235}U nucleus and produce fission (and two to three neutrons)
 - by a ^{235}U nucleus to produce non-fission reactions
 - by other materials in the reactor resulting in non-fission reactions

Key Terms

Neutron multiplication factor, k

- If exactly one neutron out of the 2 to 3 neutrons causes further fission with another ^{235}U nucleus and continues so on, a **controlled nuclear chain reaction** sets in
- In another words, if 100 neutrons undergo fission reactions with 100 ^{235}U nuclei and out of the 200 to 300 neutrons thus produced, just 100 neutrons cause fission in another 100 ^{235}U nuclei, and continuously so on, again we say a **controlled nuclear chain reaction** has set in. However, this case produces 100 times more heat energy (100 times more power)
- **Neutron multiplication factor (k) is defined as the ratio of number of neutrons in one generation to its previous generation. For both cases above $k = 1$**
- If $k=1$, reactor is “critical” (means good in nuclear engineering)
- $k < 1$, reactor is “sub-critical” and $k > 1$, reactor is “super-critical”

Key Terms

Reactivity (ρ)

- measures the degree of change in [neutron multiplication](#) (generation) in a reactor core
- is directly related to the tendency of the reactor core to change its power level
- If reactivity is positive, the core power tends to increase
- if it is negative, the core power tends to decrease
- if reactivity is zero, the core power tends to remain stable
- indicates the amount of reactivity of opposite sign that must be applied by the reactor control system in order to make a desired power level change

$$\rho = \frac{k-1}{k}$$

Technical Information for some typical NPPs

(Next 6 Slides)

Typical PWR Technical Data

Power

NSSS rated thermal output 2905 MWth

Gross electrical output 990 MWe

Reactor Core

Core rated thermal output 2895 MWth

Core damage frequency 5×10^{-6} per reactor-year

Number of fuel assemblies 157

Total weight of Uranium 72.5 t (enriched uranium)

Rod cluster control available locations (equipped with CRDMs) 61

Main Primary System

Number of reactor coolant loops 3

Number of CRDMs 61

Reactor coolant system operating pressure 155 bar (light water)

Reactor coolant temperature at RPV inlet 292.4°C

Reactor coolant temperature at RPV outlet 327.6°C

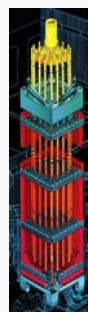
Steam pressure at SG outlet at nominal power 67.1 bar

Steam flow rate 1614 kg/s

Reactor coolant pump nominal flow 23790 m³/h

Source:

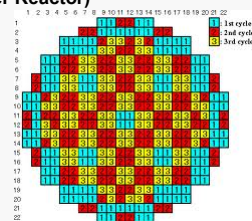
http://www.ecolo.org/documents/documents_in_english/china-LingAo-success-story.pdf



PWR

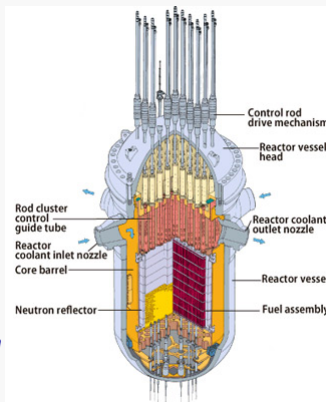
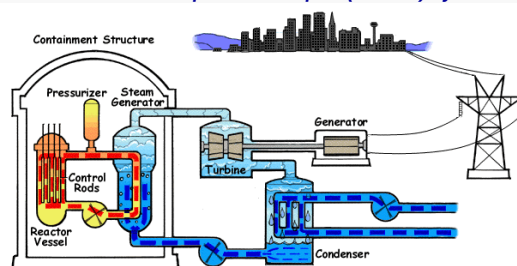
(Pressurized Water Reactor)

Typical PWR
Fuel
Assembly



Typical PWR Fuel Loading Pattern

Basic Dual Loop Heat Transport (coolant) System



Source: www.mhi.co.jp

Typical PWR Vessel

Image Courtesy:

<http://www.nucleartourist.com/images/Fuel-c.jpg>

<http://www.power-technology.com/projects/tomari-3/tomari-34.html>

http://aris.iaea.org/sonar/image/p00060_image006.gif

<http://www.nrc.gov/reading-rm/basic-ref/students/animated-pwr.html>

Pressurized Heavy Water Reactor (CANDU 6) – Technical Data

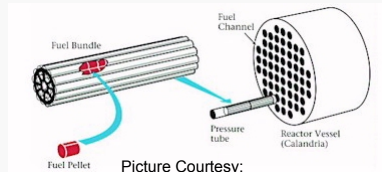
Reactor			
Type		PHWR	
Thermal output	(PHTS)	2064 MW(th)	
Coolant flow rate	(PHTS)	7.7 Mg/s	
Design temperature	(RIH)	279° C	
Design pressure	(RIH)	12.9 MPa(g)	
Operating temperature	(RIH)	266° C	
Operating pressure	(RIH)	11.75 MPa(abs)	
Design temperature	(ROH)	316° C	
Design pressure	(ROH)	10.7 MPa(g)	
Operating temperature	(ROH)	310° C	
Operating pressure	(ROH)	10.0 MPa(abs)	
Fuel Channels			
Pressure tube inside diameter		103.38 mm	
(cold, unpressurized)			
Core Length (between calandria tubesheets)		5.94 m	
Number of pressure tubes		380	
Coolant flow (nominal)		24 kg/s	
Est. pressure drop – 12 bundles		838 kPa	
Fuel			
Length of bundle		495.3 mm	
Outside dia. of bundle (over bearing pads)		102.4 mm	
Weight of bundle (nominal)		23.7 kg	
Weight of uranium per bundle (nominal)		19.2 kg	
Sheath outside dia. (cold)		13.1 mm	
Sheath thickness (average)		0.4 mm	
Sheath material		Zircaloy - 4	
Elements per bundle		37	
Fuel material		Natural UO ₂	
Fuel bundles in core		4560	
Fuel bundles per channel		12	
Heat Transport System			
Number of loops		2	
Primary coolant		D ₂ O	
Reactor inlet temperature		266° C	
Reactor outlet temperature		310° C	
Reactivity Control Units			
Power control		14 light water compartments	
		21 stainless steel rods	
		4 cadmium rods	
		28 cadmium rods (vertical)	
		6 gadolinium nitrate injection tubes (horizontal)	
Safety shutdown			

Source:

<http://www.aec.ca/Assets/Publications/C6-Technical-Summary.pdf>

CANDU Reactor

(‘CAN’ada Deuterium Uranium)



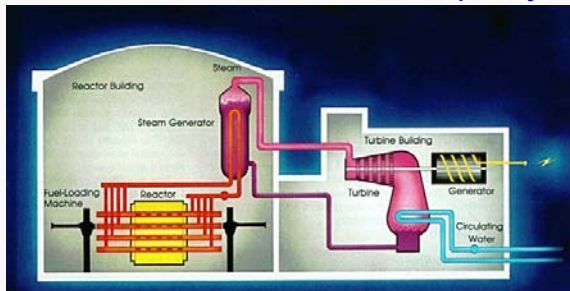
Picture Courtesy:
[Atomic Energy of Canada Limited](http://www.aec.ca)

(CANDU Fuel Bundles)

The copyright holder of this file, [Atomic Energy of Canada Limited](http://www.aec.ca), allows anyone to use it **for any purpose**, provided that the copyright holder is properly attributed



(Horizontal Pressure Tubes in Heat Transport System)



Source:
<http://www.nucleartourist.com/images/candu2.jpg>

(Calandria Vessel)

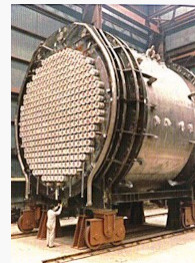


Photo Courtesy:
[Atomic Energy of Canada Limited](http://www.aec.ca)

Typical FBR – Technical Data

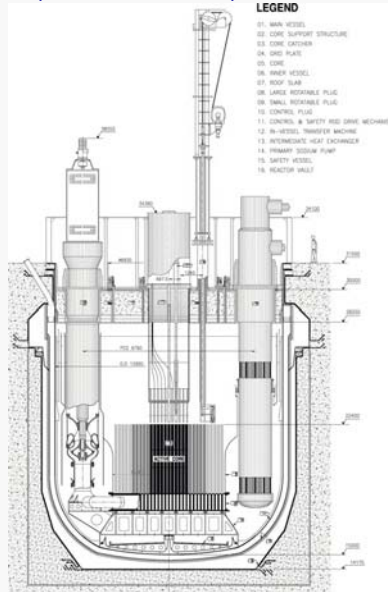
Thermal power (MWt)	1250
Electric output (MWe)	500
Core height (mm)	1000
Core diameter (mm)	1900
Fuel (MOX)	$\text{PuO}_2\text{-UO}_2$
Fuel pin outer diameter (mm)	6.6
Pins per fuel subassembly	217
Fuel clad material	20% CW D9
Diameter of main vessel (mm)	12900
Primary circuit layout	Pool
Primary Sodium inlet/outlet temp (°C)	397/547
Steam temperature (°C)	490
Steam pressure (MPa)	16.6
Reactor containment	Rectangular
Plant life (years)	40
No. of shutdown systems	2
No. of decay heat removal systems	2

Source:

"The design of the Prototype Fast Breeder Reactor", S.C. Chetal et.al., Nuclear Engineering and Design 236 (2006) 852-860

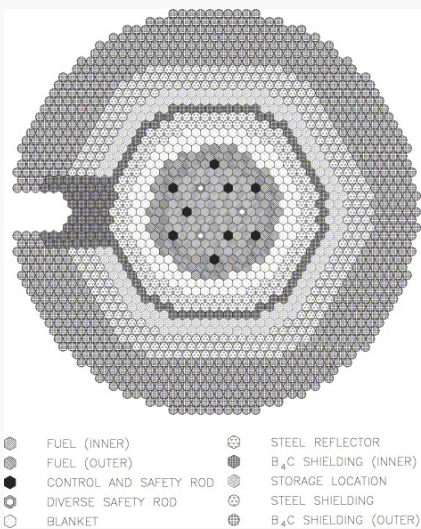
FBR

(Fast Breeder Reactor)



Source:

"The design of the Prototype Fast Breeder Reactor", S.C. Chetal et.al., Nuclear Engineering and Design 236 (2006) 852-860



References

1. International Atomic Energy Agency, "Manual of Acid in Situ Leach Uranium Mine Technology," IAEA-TECDOC-1239, Vienna, 283pp, 2001
2. International Atomic Energy Agency, "Report on the preliminary fact-finding mission following the accident at the nuclear fuel processing facility in Tokaimura, Japan," 1999
3. International Atomic Energy Agency, "Safety of Conversion Facilities and Uranium Enrichment Facilities," Specific Safety Guide No. SSG-5, IAEA, Vienna, 2010
4. O. Pederson, "Developments in the Uranium Enrichment Industry," IAEA Bulletin - Vol.19, No. 1, IAEA, Vienna, 1977
5. A.S. Krass, P. Boskma, B. Elzen and W.A. Smit. Uranium Enrichment and Nuclear Weapon Proliferation, Taylor and Francis Ltd, London and New York, 1983
6. R.R. Judkins and A.R. Olsen, "Estimation of Costs for Fabrication of Pressurized-Water Reactor Fuel," ORNL/TM-6501, January 1979
7. M. Ragheb, "Isotopic Separation and Enrichment," Course notes, <https://netfiles.uiuc.edu/mragheb/www/NPRE%20402%20ME%20405%20Nuclear%20Power%20Engineering/Isotopic%20Separation%20and%20Enrichment.pdf>
8. W. Charlton and D. Boyle, Course Presentation Slides, Private Communication

List of Required Texts

- The following textbook is required:
 - R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management, 2nd Edition*, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999
- This text is available for purchase online at <http://www.ans.org/store/vi-350015>
- Readings for the lectures and selected problem sets will be assigned from this text

Assignments, Activities, Exercises

- Reading Assignments
 - C&T Ch. 4, 6
 - Assorted papers
- Exercises
 - Ragheb Ch. 10 problems 1-4
- Activity
 - Why not lump uranium in the core instead of having pins and assemblies?

End of Lecture 5

Power Operations - 2

Lecture 6

GNEII Pilot Course

February – May 2011

Khalifa University of Science, Technology and Research

Abu Dhabi, United Arab Emirates

William Charlton

Texas A&M University, USA

SAND 2010_____

Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

Power Operations - 2

- **Main Teaching Points**

1. Fuel behavior during operation and its effect on operation
2. The fuel loading design – fissile loading requirements, fuel cycle length, power profile design, etc.

Power Operations - 2

- **Key Terms** – batch cores, uranium decay chain, thorium decay chain, fission product yield, core reactivity, burnup, chemical shim, absorber rods, in core fuel management, refueling scheme, neutron flux
- **Key Concepts** – fuel material changes in core, benefits of batch refueling, purpose of chemical shim and neutron absorber rods, understand optimization parameters for in core fuel management

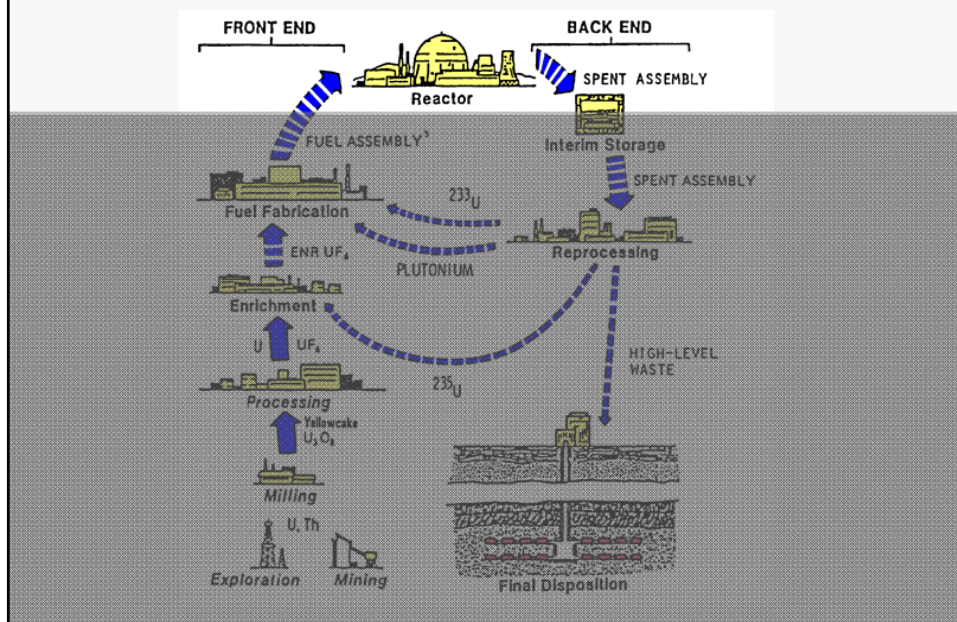
Power Operations - 2

- **Desired Student Outcomes:**
 1. Understand fuel material changes in core
 2. Knowledge about the benefits of batch refueling
 3. Be familiar with the purpose of chemical shim and neutron absorber rods
 4. Understand optimization parameters for in core fuel management
 5. Understand the purpose of and methods used for reshuffling and reloading

Assumptions and Prerequisites

- Successful completion of GNEII weeks one and two
- Bachelor's degree
- College level math

Nuclear Fuel Cycle



Initial Core Fuel Loading

- When a nuclear reactor is loaded initially, it is loaded with **fresh fuel assemblies** everywhere in the core
- This will lead to a center peaked power distribution and a **high reactivity** (more than what is required to maintain a controlled chain reaction)
- Initial high reactivity and power distributions are controlled using **neutron absorbers**
 - **Neutron absorber rods in fuel assembly**
 - **Neutron absorbing material in coolant water (soluble boron)**
- As we operate the reactor, the power distribution from fission will change due to **change in the fuel isotopic concentrations**
- To properly manage the core, we will need to determine the best location for each fuel assembly for each operating fuel cycle
 - minimize the radial power peaking factor,
 - maximize fuel burnup, and
 - minimize soluble boron concentrations

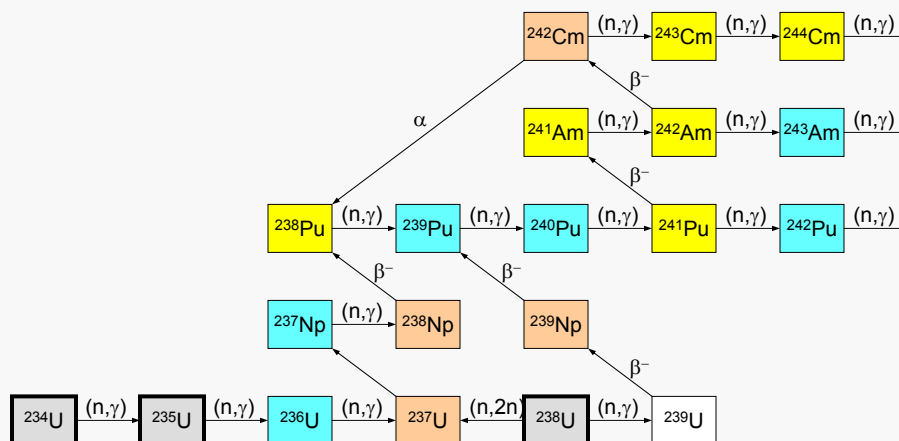
Changes in Fuel Composition

- The initial fuel in a reactor essentially consists of only ^{234}U , ^{235}U , and ^{238}U in a LWR where as MOX in a FBR has plutonium as well
- As the reactor is operated the fuel composition begins to change
 - some neutrons cause fission reactions which result in the production of **fission products**
 - these fission products might then decay to other isotopes
 - some neutrons are absorbed through (n,gamma) reactions in U isotopes and produces **higher mass U isotopes**
 - these higher mass U isotopes might then decay by Beta⁻ emission to **higher actinide isotopes** (Np, Pu, Am, Cm, etc.)

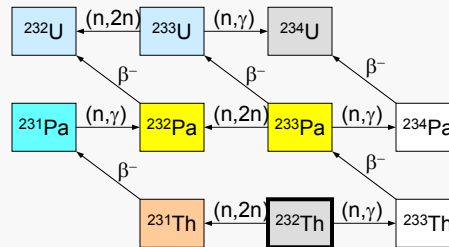
Reactions Resulting in Material Changes

- Scattering reactions (elastic and inelastic) leave the materials unchanged
- All other reactions result in material changes:
 - radiative capture
 - fission
 - (n,2n), (n,3n), etc..
 - (n,p), (n,d), (n,t), (n, α), etc..
- Some of these isotopes can then decay to new isotopes or absorb neutrons and form new isotopes

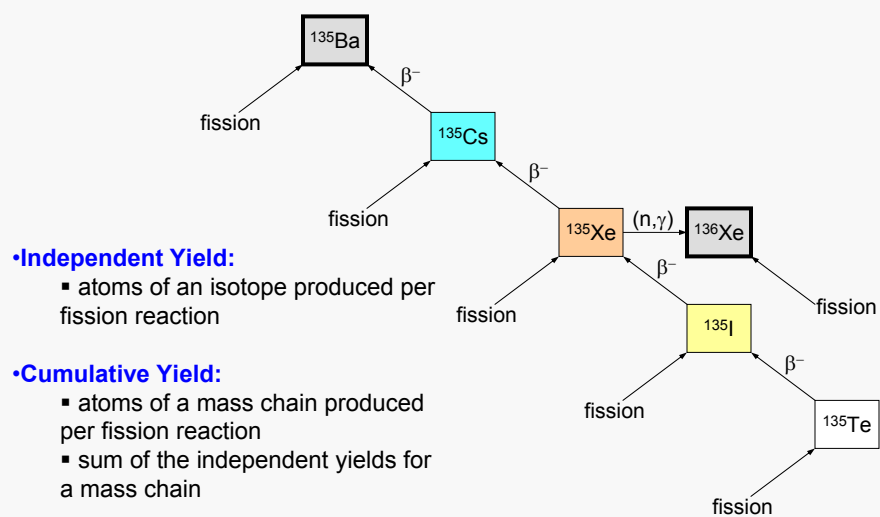
Uranium Buildup and Decay Chain



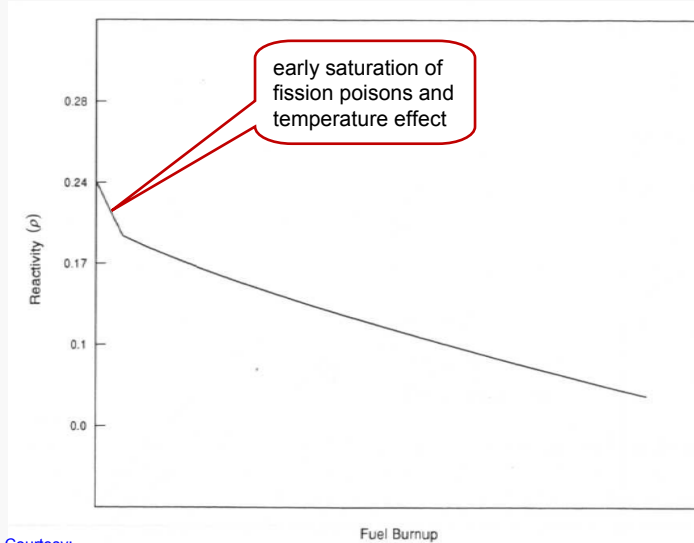
Thorium Buildup and Decay Chain



Fission Product Buildup and Decay Chain



Core Reactivity Versus Burnup

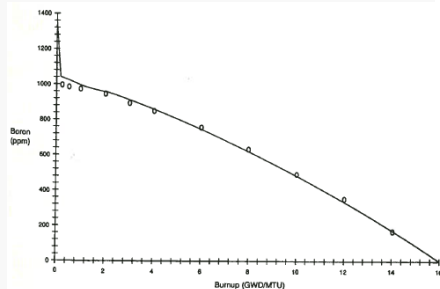


Picture Courtesy:

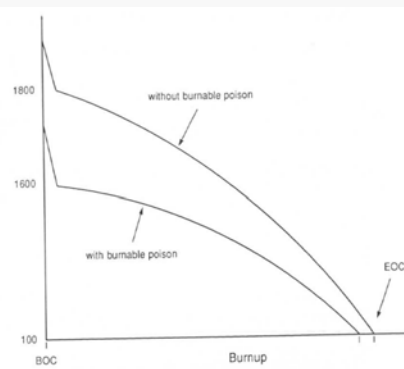
R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management*, 2nd Edition, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

Chemical Shims and Burnable Absorbers

- To offset the high initial reactivity of the core we use
 - soluble boron
 - burnable absorbers



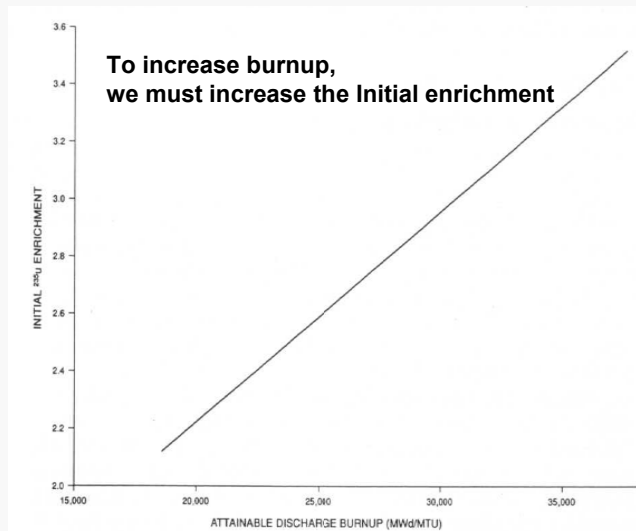
The change of boron concentration versus burnup in a typical PWR.



Picture Courtesy:

R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management*, 2nd Edition, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

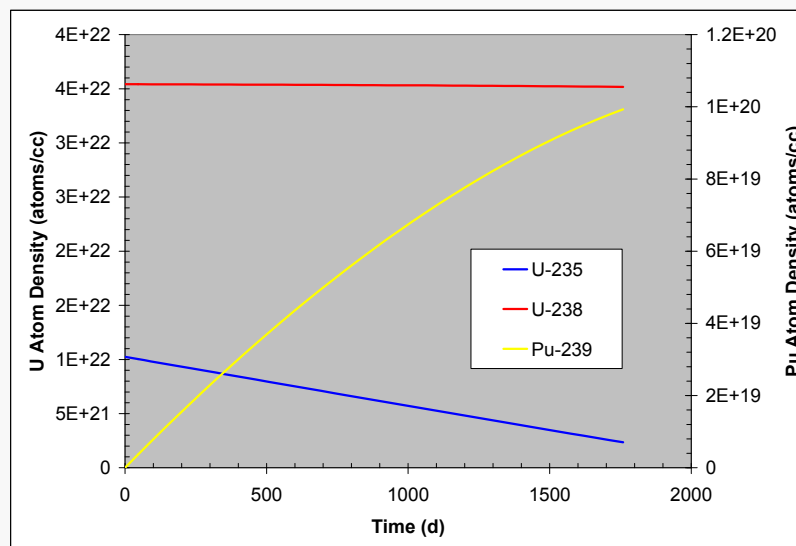
Final Burnup Versus Initial Enrichment



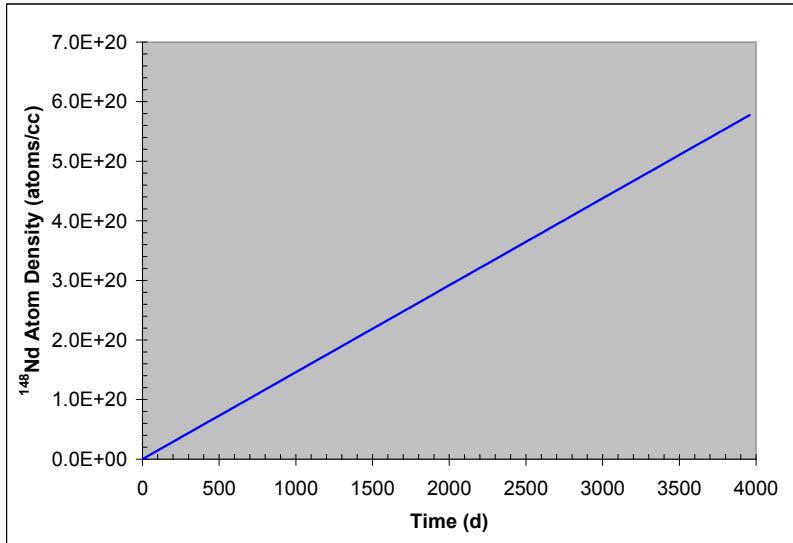
Picture Courtesy:

R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management, 2nd Edition*, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

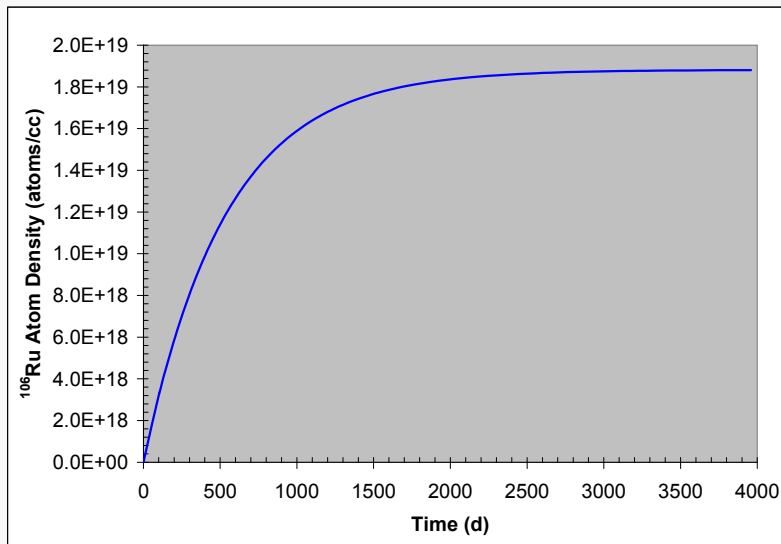
Major Actinide Buildup and Decay



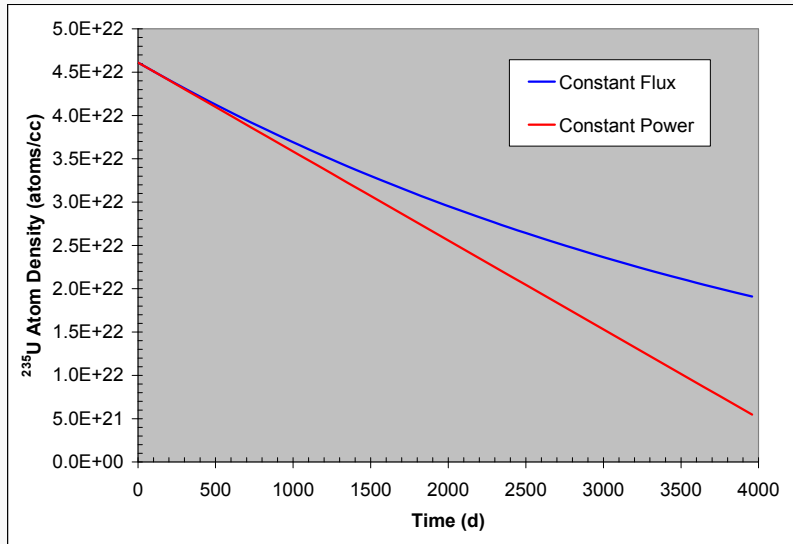
^{148}Nd Fission Product Buildup



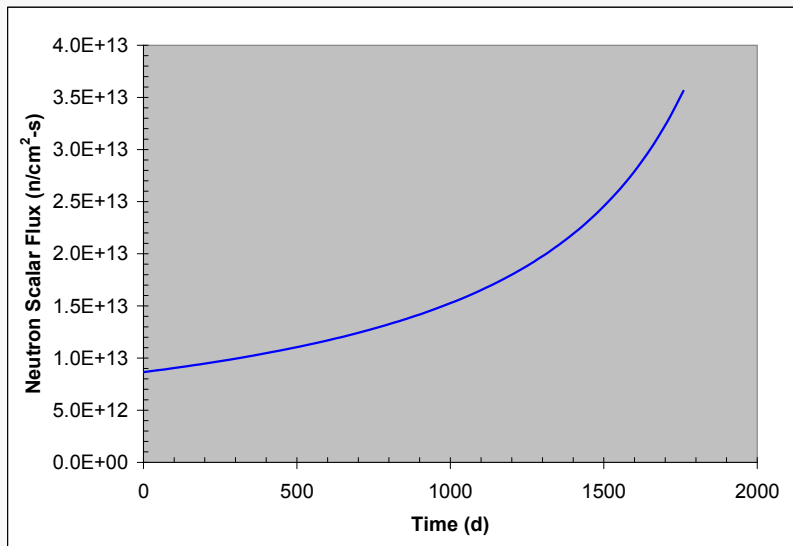
^{106}Ru Fission Product Buildup



Solution for ^{235}U with Two Conditions



Neutron Scalar Flux Versus Time



In Core Fuel Management

Understand how the multi-batch core affects fuel management operations

Illustrate in the following slides

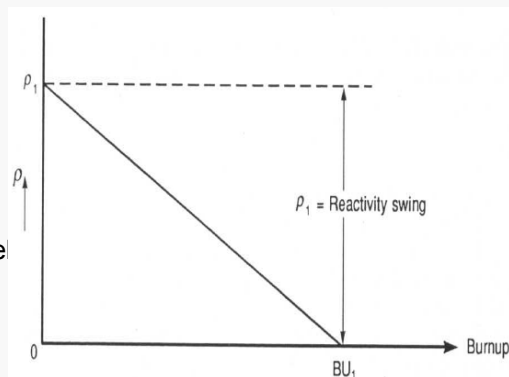
one batch core refueling

two batch core refueling

three batch core refueling

One Batch Core

- Fresh core has a reactivity swing
- The reactivity swing in the beginning has to be offset by chemical shims, burnable absorbers, and control rod movements
- The reactivity of the fuel is decreasing with the decrease in ^{235}U concentration
- Figure shows the reactivity of the core starting from fresh fuel loading to the end where all fuel assemblies are replaced with fresh ones (one batch core)

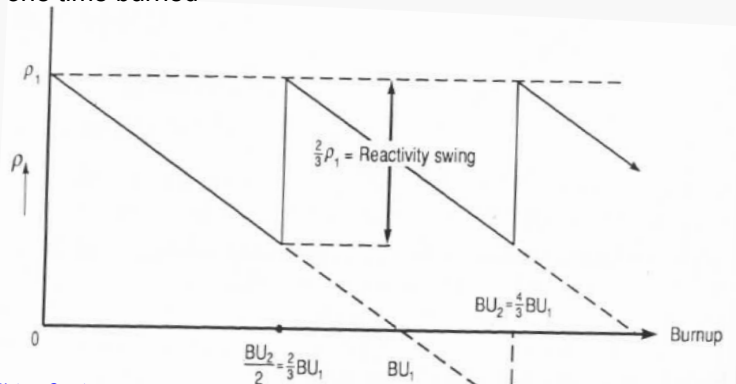


Picture Courtesy:

R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management*, 2nd Edition, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

Two Batch Core

- Figure shows the reactivity of the core starting from fresh fuel loading and then after half of the fuel assemblies are replaced to increase reactivity
- So after first refueling, half the core will be fresh and next half will be one time burned



Picture Courtesy:

R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management*, 2nd Edition, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

Three Batch Core

- Load fresh fuel in the entire core
- After first fuel cycle, replace one third of the burned (once burned) assemblies with fresh fuel
- After second fuel cycle, replace one third of the burned (twice burned) assemblies with fresh fuel
 - So after second refueling, core will have 1/3 fresh fuel, 1/3 once burned and 1/3 twice burned
- At the end of third fuel cycle, replace thrice burned assemblies with fresh fuel
- Then on the process is repeated ([equilibrium cycle](#)), that is always replace thrice burned fuel assemblies with fresh assemblies
- Three batch core is usually followed in PWRs

Why Not Use Many Batches?

- So for a PWR, why don't I add in many small batches?
- How many batches do we typically use in a PWR?
- How about for a CANDU?

Core Reloading Pattern

b_{ij} ; i = batch number and
 j = number of cycles irradiated

TABLE 6.2
 RELOADING PATTERN FOR A THREE-BATCH CORE

Batches in Core	1	2	Cycle 3	4	5
1	$b_{1,0}$				
2	$b_{2,0}$	$b_{2,1}$			
3	$b_{3,0}$	$b_{3,1}$	$b_{3,2}$		
4		$b_{4,0}$	$b_{4,1}$	$b_{4,2}$	
5			$b_{5,0}$	$b_{5,1}$	$b_{5,2}$
6				$b_{6,0}$	$b_{6,1}$

Table Courtesy:

R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management*, 2nd Edition, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

Refueling Scheme

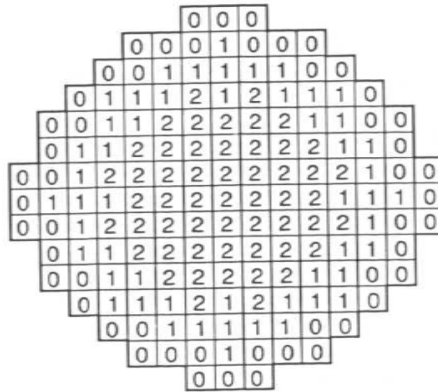


Fig. 6.7 The OUT-IN refueling scheme. The core is divided into three regions: 0 = fresh fuel; 1 = fuel burned in one previous cycle; and 2 = fuel burned in two previous cycles.

Picture Courtesy:

R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management, 2nd Edition*, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

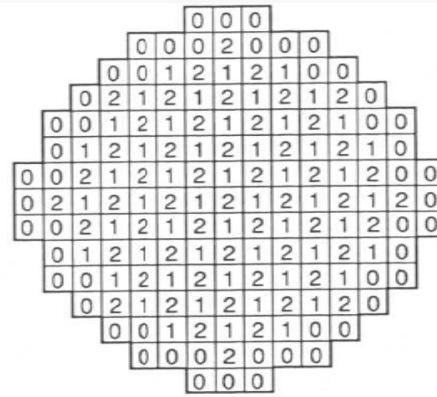
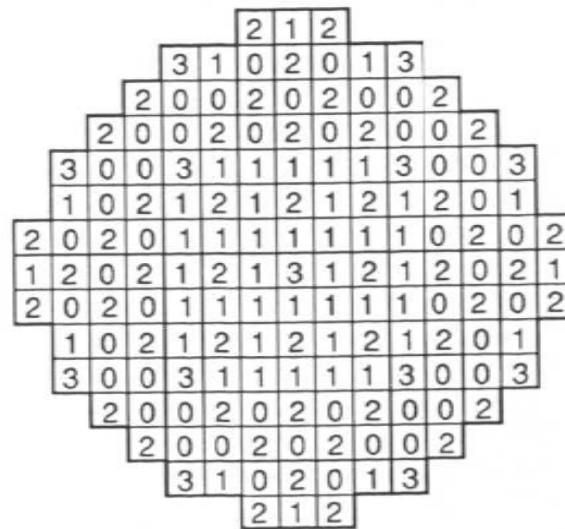


Fig. 6.8 The SCATTER (checkerboard) refueling scheme: 0 = fresh fuel; 1 = fuel burned in one previous cycle; and 2 = fuel burned in two previous cycles.

Low-Leakage Refueling Scheme



Picture Courtesy:

R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management, 2nd Edition*, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

In-Core Fuel Management Calculations

- So within the loading pattern for each batch, how do we determine where to put each assembly?
 - Lattice Physics Calculation (transport)
 - Full Core Model (diffusion)
 - Fuel Burnup Calculation (buildup and depletion)
 - Repeat
 - Optimization Module (conjugate gradient, simulated annealing, etc.)

81

References

1. International Atomic Energy Agency, "Manual of Acid in Situ Leach Uranium Mine Technology," IAEA-TECDOC-1239, Vienna, 283pp, 2001
2. International Atomic Energy Agency, "Report on the preliminary fact-finding mission following the accident at the nuclear fuel processing facility in Tokaimura, Japan," 1999
3. International Atomic Energy Agency, "Safety of Conversion Facilities and Uranium Enrichment Facilities," Specific Safety Guide No. SSG-5, IAEA, Vienna, 2010
4. O. Pederson, "Developments in the Uranium Enrichment Industry," IAEA Bulletin - Vol.19, No. 1, IAEA, Vienna, 1977
5. A.S. Krass, P. Boskma, B. Elzen and W.A. Smit. Uranium Enrichment and Nuclear Weapon Proliferation, Taylor and Francis Ltd, London and New York, 1983
6. R.R. Judkins and A.R. Olsen, "Estimation of Costs for Fabrication of Pressurized-Water Reactor Fuel," ORNL/TM-6501, January 1979
7. M. Ragheb, "Isotopic Separation and Enrichment," Course notes, <https://netfiles.uiuc.edu/mragheb/www/NPRE%20402%20ME%20405%20Nuclear%20Power%20Engineering/Isotopic%20Separation%20and%20Enrichment.pdf>
8. W. Charlton and D. Boyle, Course Presentation Slides, Private Communication

82

List of Required Texts

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- This text is available for purchase online at <http://www.ans.org/store/vi-350015>
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83

Assignments, Activities, Exercises

- Reading Assignments
 - C&T Ch. 4, 6
 - Assorted papers
- Exercises
 - Ragheb Ch. 10 problems 1-4
- Activity
 - Develop the costs associated with the acquisition of nuclear fuel. Consider the costs associated with the acquisition of ore, mining, milling and conversion, as well as enrichment for a standardized PWR

84

Assignments, Activities, Exercises

- Reading Assignments
 - C&T Ch. 4, 6
 - Assorted papers
- Exercises
 - Ragheb Ch. 10 problems 1-4
- Activity
 - Why not lump uranium in the core instead of having pins and assemblies?

End of Lecture 6

Week 3 - Day 3

Nuclear Fuel Cycle

Lecture 7 – Spent Fuel Storage

Lecture 8 – Irradiated Fuel Measurements

Lecture 9 – Spent Fuel Recycling

Spent Fuel Storage Lecture 7

GNEII Pilot Course

February – May 2011

**Khalifa University of Science, Technology and Research,
United Arab Emirates, Abu Dhabi**

(William Charlton)

Texas A&M University, USA

SAND 2010_____

Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

Spent Fuel Storage

- **Main Teaching Points**

1. Fuel handling operations
2. Need for interim spent fuel storage
3. Temporary storage options
4. Wet and dry storage options
5. Requirements of wet and dry storage
6. Spent fuel transport

Spent Fuel Storage

- **Key Terms** – spent fuel pool, wet storage, dry storage, spent fuel transport, spent fuel handling
- **Key Concepts** – Need for spent fuel assemblies interim cooling, dry casks are used when pools are full, special designed of wet and dry storage to prevent criticality, spent fuel transport methods

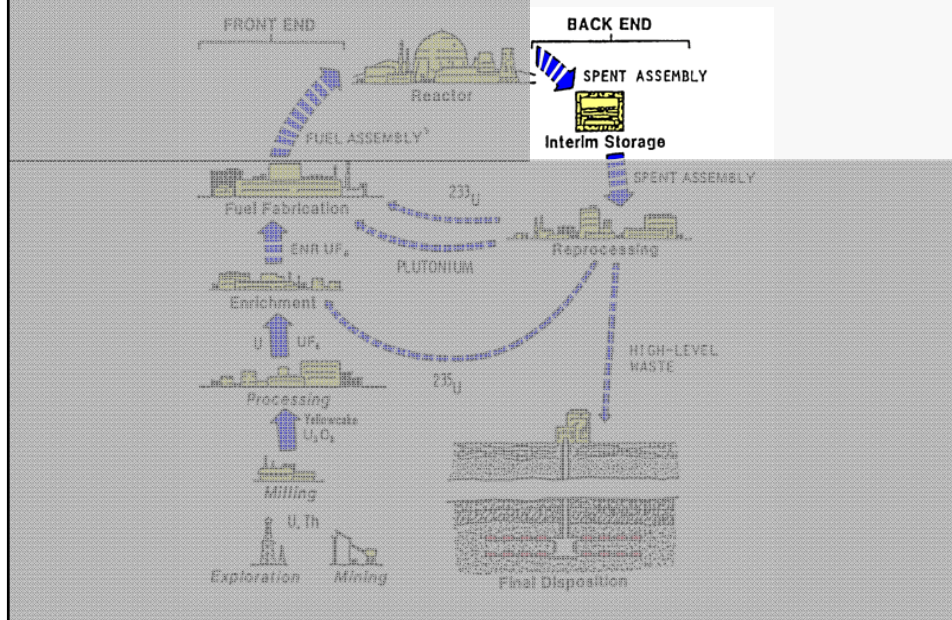
Spent Fuel Storage

- **Desired Student Outcomes:**
 1. Know why spent fuel pools are used and how the water acts as shielding
 2. Understand the reasons dry casks are used on cooler fuel and how the specially designed features make it resistant to damage and corrosion
 3. Be able to describe some of the safety features of casks that allow fuel to be safely transported with regard to criticality and proliferation concerns
 4. Be familiar with the spent fuel storage methods
 5. Strategies for spent fuel storage
 6. Be familiar with spent fuel transportation

Assumptions and Prerequisites

- Successful completion of GNEII weeks one and two
- Bachelor's degree
- College level math

Nuclear Fuel Cycle

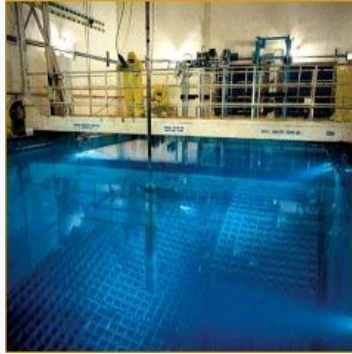


Cutaway View of PWR Internal along with Spent Fuel Pool Storage (No. 16)

LEGEND:		Estimated Duration (Hours)	
1	CONTAINMENT		
2	FUEL BUILDING		
3	REFUELING CAVITY		
4	FUEL TRANSFER CANAL		
5	OPERATING DECK		
6	CONTAINMENT POLAR CRANE		
7	REACTOR VESSEL HEAD ASSEMBLY (ON STORAGE STAND)		
8	UPPER INTERNALS STRUCTURE (ON STORAGE STAND)		
9	LOWER INTERNALS STORAGE STAND		
10	REACTOR VESSEL		
11	REACTOR CORE		
12	REFUELING MACHINE		
13	ROD CLUSTER CONTROL CHANGING FIXTURE		
14	FUEL TRANSFER TUBE		
15	FUEL HANDLING MACHINE AND HOIST		
16	FUEL STORAGE RACKS		
17	SPENT FUEL CASK LOADING AREA		
Operation		Estimated Duration (Hours)	
Remove CRDM fans and cooling ducts **		5	
Remove missile shield **		2	
Remove RV head insulation		4	
Remove seismic tie rods		2	
Disconnect CRDM cables **		6	
Disconnect thermocouples		4	
Remove cable tray and missile shield **			
support beams		6	
Detension and remove studs **		31 (9)*	
Install guide studs and plugs		2	
Install head lift rig **		2	
Remove RV head assembly		4	
Disconnect control rod drive shafts		6	
Remove upper internals		4	
Fuel shuffle		83	
Reconnect control rod drive shafts		11	
Replace RV head assembly		4	
Remove head lift rig		2	
Remove guide studs and plugs		2	
Install and tension studs		45 (11)*	
Connect cable tray and missile **			
shield support beams		4	
Connect thermocouples		10	
Connect CRDM cables **		10	
Connect seismic tie rods		4	
Install RV head insulation		4	
Install missile shield **		4	
Install CRDM fans and duct work**		5	
Totals		266	

Source: "The Westinghouse Pressurized Water Reactor Nuclear Power Plant," Westinghouse Electric Corporation, Pittsburgh, Pennsylvania, pg. 184 (1984).

PWR Fuel Handling in Spent Fuel Pool and Inside Containment



Spent Fuel Pool

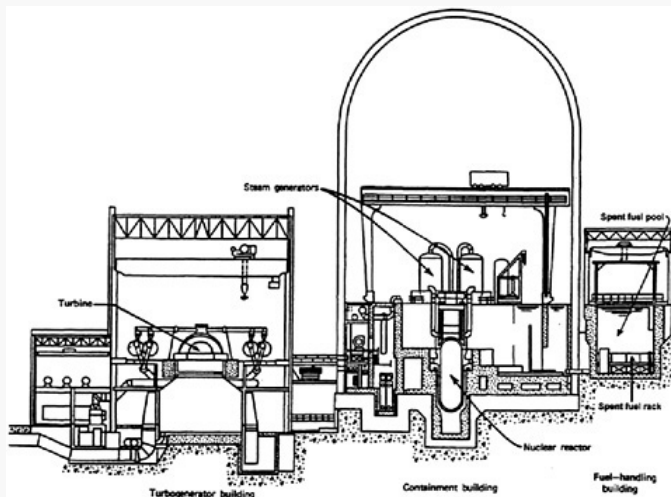
Photo Courtesy:
<http://www.solcomhouse.com/nuclearpowerplants.htm>



Open PWR Pressure Vessel for Fuel Handling Operations

Photo Courtesy:
<http://www.nuclearcounterfeit.com/?tag=pwr>

PWR Containment and Fuel-Handling Building



Schematic section through a PWR reactor plant. The spent fuel pool is located in the fuel-handling building next to the domed reactor containment building at or slightly below ground level,

SOURCE: Modified from Duderstadt and Hamilton (1976, Figure 3-4).

Spent Fuel Storage in Open Fuel Cycle

- Spent fuel discharged from reactor is cooled for about 10 to 20 years in a large water pool (**wet storage**)
- During wet storage fission products decay and radiation intensity reduces
- The 10 to 20 years cooled fuel assemblies are transferred into dry casks for long term storage (**dry storage**)



Onsite spent fuel pool (**wet storage**)
Picture courtesy: <http://world-nuclear.org>



Spent fuel dry storage in casks
Picture Courtesy: USNRC

Spent Fuel Storage in Closed Fuel Cycle

- Spent fuel discharged from reactor is cooled for about 10 to 20 years in a large water pool (**wet storage**)
- Spent fuel undergoes **reprocessing** (chemical process) to separate fission products from depleted uranium and plutonium
- Separated uranium oxide and plutonium oxide is recycled as mixed oxide (**MOX**) fuel in reactors
- Separated fission products undergoes **high level waste management**

Spent Fuel Decay Heat

- Decay heat from spent fuel as function of time is important to
 - Cool the core adequately after a normal or abnormal shutdown (LOCA)
 - Know the cooling requirements for the fuel during storage and transportation
 - Dispose the fuel in repository
- Decay heat measurements
 - Measurements using **radiometry** (based on the measurement of the energy spectrum of betas and gammas emitted by the FP) or **calorimeter** (uses a large calorimeter where all the energy released by the FP is absorbed and measured)

At Reactor Wet Storage Requirements

- Intensely radioactive freshly-discharged fuel (4-10 ft thick concrete; water)
- Subcriticality
- Thermal management
- Provisions needed to receive fuel from the reactor
- Positive fuel bundle location and accurate record keeping
- Water chemistry and pool leak tightness
- Ventilation for humidity and airborne contaminants
- Special ops (e.g., leaking fuel elements, re-use of fuel)

Typical Wet Storage Requirements

- **Stainless Steel (SS) storage racks with 6" or more of water between fuel assemblies**
- **SS replaced with Boral (a neutron-absorbing composite material consisting of boron carbide evenly dispersed within a aluminum matrix)**
 - Permitted close spacing of fuel assemblies without criticality problems
 - Boron absorbs thermal neutrons without producing significant secondary radiation
 - Only alpha particles, which is easily absorbed by water, and a soft 0.48-MeV gamma emitted
- **Basic requirement of any spent fuel storage facility is that the neutron multiplication factor, $k < 1.0$**
 - By appropriate distance between fuel assemblies
 - Use of solid neutron poisons (eg., Boral) and no dissolved boron in the water pool

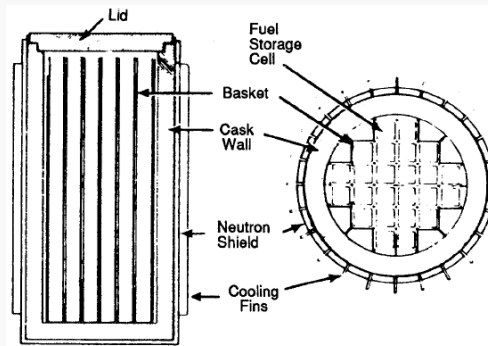
Other Storage Options

- **Dry storage**
 - Store in a canister that is cooled by a gas
 - Adequate radiation shielding
 - Need not be on-site (at reactor , AR) storage and can be off-site (away from reactor, AFR)
 - NRC approved designs; NAC-128, MC-10, TN-24. VSC, NAC-STC
 - Assumed the stored irradiated fuel having composition of fresh fuel without giving any burnup credit and burnup credit is under consideration
 - **Dose rate at boundary should not exceed 25 mrem/h(0.25mSv/h)**
- **Spent fuel rod consolidation**
 - "unconsolidated" fuel is when fuel is stored as intact assemblies
 - "consolidated" fuel - fuel rods are removed from the assembly and placed in a grip with closer spacing or in a closed-packed array inside a canister
 - Scrap radioactive structural parts should be stored properly

Dry Storage

➤ Cylindrical metal cask

- 8ft diameter and 16ft long
- 21 to 33 PWR assemblies (~9 to 14 t of heavy metal)
- 45 to 70 BWR assemblies (~8 to 12 t of heavy metal)
- Fully loaded weight ~ 100 to 120 t
- Walls made of iron or iron plus lead with adequate thickness to shield gamma rays
- Additional neutron absorbing material (polyethylene or resin)
- Fuel baskets made of combinations of steel, Al, Cu and B)
- External surface may have fins to enhance cooling
- 25ft² pad area for vertical placement



The design of a typical metal cask for dry storage of spent fuel

Image Courtesy:

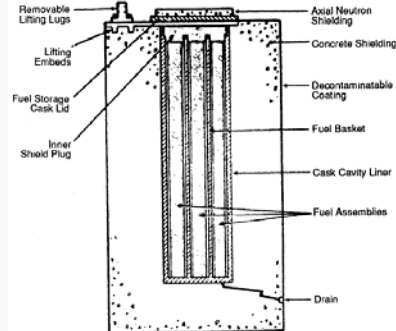
R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management*, 2nd Edition, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

Dry Storage (cont'd)

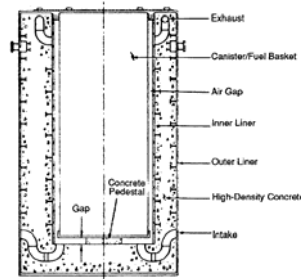
- Dual-purpose metal cask
- For storage as well as transport
- Must satisfy both storage and transport criteria
- A little more costly than the single purpose metal cask

Dry Storage Concrete Cask

- Heavily reinforced concrete (~3 ft thick concrete); ventilated/unventilated
- Adequate gamma-ray shielding and a steel inner liner for concrete
- ~8.5 ft diameter and 18 ft long (unventilated-loaded weighs ~90t)
- ~12 ft diameter and 20 ft long (ventilated- loaded weighs ~125t)
- ~24 PWR assemblies in He atmosphere
- Decay heat removed by radiation/conduction/convection
- Concrete vaults



The design of an unventilated concrete cask for dry storage of spent fuel

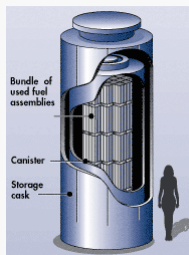


The design of a ventilated concrete cask for dry storage of spent fuel

Picture Courtesy:

R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management*, 2nd Edition, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

Dry Storage Designs



Schematic Courtesy:

<http://www.nrc.gov/waste/spent-fuel-storage/diagram-typical-dry-cask-system.html>



A typical small SNF shipping cask being mounted on a truck

Photo Courtesy:

http://en.wikipedia.org/wiki/Spent_nuclear_fuel_shipping_cask

After 10 or more years, the radiation and decay heat levels are low so as to store fuel assemblies in large casks, cooled by air passing on the outside. Such casks typically hold 24 to 40 fuel assemblies. The casks have double metal ring seals and are bolted to ensure radioactive material release should not be able to occur. The casks typically are 17 feet high and 9 feet across. Pressurized helium gas is used inside the cask with a to promote heat removal from the fuel assemblies to the cask wall. Air on the outside removes the heat.

Dry Storage Pros and Cons

- Avoids corrosion and leak potential of water pools
- Less moderation provides high density packing (8-14 metric ton heavy metal [MTHM])
- Shielding and thermal management more difficult
- Good for long term storage after initial cool down
- Secure and safe (typically weighs 100+ tons)
- Ventilated and unventilated designs

Spent Fuel Transport

- Transport casks
 - Lighter casks , 25 to 40 t, holds one to seven assemblies and can be carried by trucks
 - Heavier casks, up to 120 t, can carry up to 36 assemblies and can be transported by rail
- Transport cask example
 - Fuel assemblies are sealed in a water-filled SS cylinder with walls 0.5" thick, clad with 4" of a heavy metal, usually lead, for radiation shielding
 - Further surrounded by 5" of water and encircled by a corrugated SS outer package
- Transportation cask tests
 - Crash tests; Locomotive tests; Crash fire test; Drop test; etc.



Courtesy Pacific Nuclear

References

1. International Atomic Energy Agency, "Spent fuel performance assessment and research," IAEA-TECDOC-1343, Spent Fuel Performance Assessment and Research (SPAR) 1997–2001, Vienna, 2003
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3. T.L. Bridges, A.L. Lengyel, D.K. Morton, and D.L. Pincock, "Standardized DOE Spent Nuclear Fuel Canister And Transportation System For Shipment To The National Repository," INEEL/CON-00-01229 PREPRINT, February 2001
4. International Atomic Energy Agency, "Fuel failure in water reactors: Causes and mitigation," IAEA-TECDOC-1345, Vienna, March 2003
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http://www.armscontrolcenter.org/assets/ppt/vonhippel_PP.ppt
6. World Nuclear Association, "Processing of Used Nuclear Fuel," <http://www.world-nuclear.org/>
7. J. Cerami, "Nuclear Fuel Reprocessing: Now We're Ready(?)," Presentation Slides, <http://www.kentlaw.edu/faculty/fbosselman/classes/EnergyLawSp07/PowerPoints/CeramiNuclearFuelReprocessing.ppt>
8. W. Charlton and D. Boyle, Course Presentation Slides, Private Communication

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Reading Assignments

- C&T Ch. 7, 8, 9
- Assorted papers

Exercises

- C&T problems 6.2, 6.6
- Develop a position paper associated with storing spent fuel vs pursuing a reprocessing strategy (consider both an indigenous or external reprocessing capability). What aspects drive the costs? What political ramifications are there? Legal aspects?

End of Lecture 7

Irradiated Fuel Measurements Lecture 8

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Irradiated Fuel Measurements

- **Main Teaching Points**

1. How fuel changes under irradiation
2. Types of fuel measurements
3. Important fission products associated with measurements and storage

Irradiated Fuel Measurements

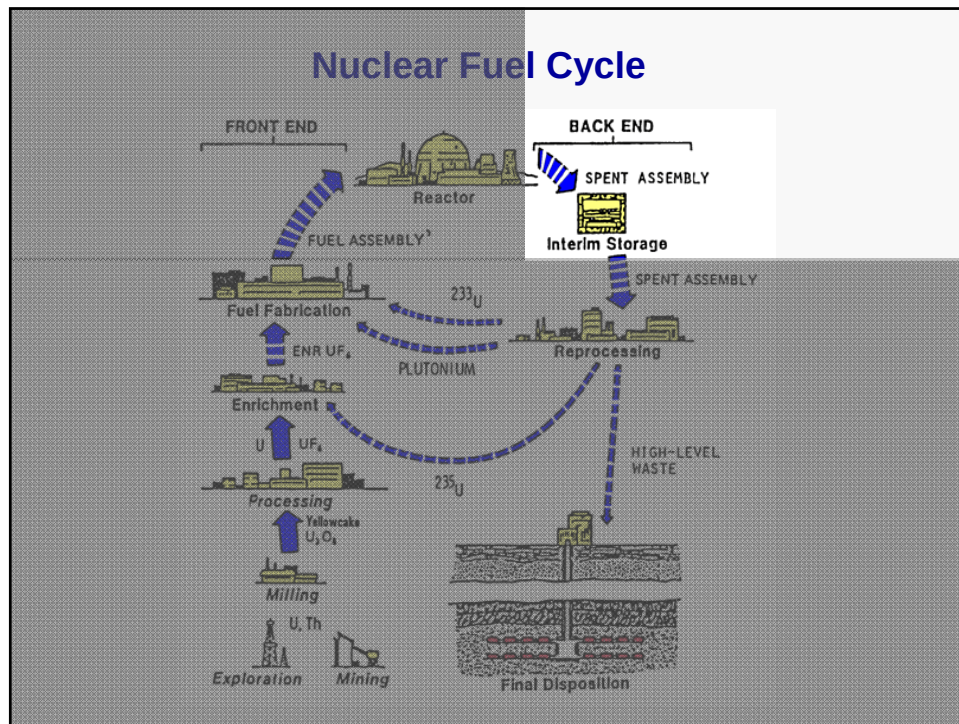
- **Key Terms** – fission products, actinides, activation products, fission product yield, burnup measurement, fork detector
- **Key Concepts** – fission product inventory in spent fuel, non-destructive assay for spent fuel burnup

Irradiated Fuel Measurements

- **Desired Student Outcomes:**
 1. Understand about spent fuel contents
 2. Familiarity with different non destructive assay (NDA) methods for spent fuel burnup while in storage

Assumptions and Prerequisites

- Successful completion of GNEII weeks one and two
- Bachelor's degree
- College level math



Radioactivity from Spent Fuel

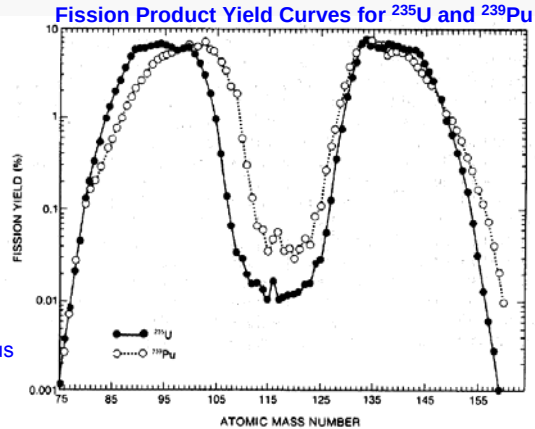
- **3 general groups of radioisotopes**
 - Fission products (~350 nuclides, very many with short half lives)
 - Actinides
 - Activation products
- **Quantities of these depend on**
 - Fuel enrichment
 - Irradiation history
 - Time after discharge

FP Inventory Depends on Fuel Fissile Isotope

(PWR average burnup (fuel exposure) is 33 GWd/tU and some PWRs operate with average fuel burnup of 45 GWD/t)

Fuel Exposure (GWd/tU)	²³⁵ U	²³⁸ U	²³⁹ Pu	²⁴¹ Pu
0	100.0	0.0	0.0	0.0
1.2	88.1	6.7	5.1	0.01
4.7	76.3	6.9	16.4	0.35
9.9	60.6	7.4	29.6	2.4
15.1	49.5	7.8	37.5	5.3
20.0	41.1	8.1	42.8	8.0
25.6	32.9	8.5	47.3	11.3
29.6	27.6	8.8	40.9	13.7
34.7	23.8	9.9	58.2	18.1
40.0	16.7	9.3	55.3	18.7
46.8	11.7	9.6	57.5	21.2

Percentage of Total Fissions Versus
Exposure for LEU PWR Fuel



Long-lived Fission Products

To obtain activity in curies/ t U, multiply by 3 (because assumption made is that the fuel stayed in the core for 3 yrs.) & divide by 81.8, the mass of fuel in tons

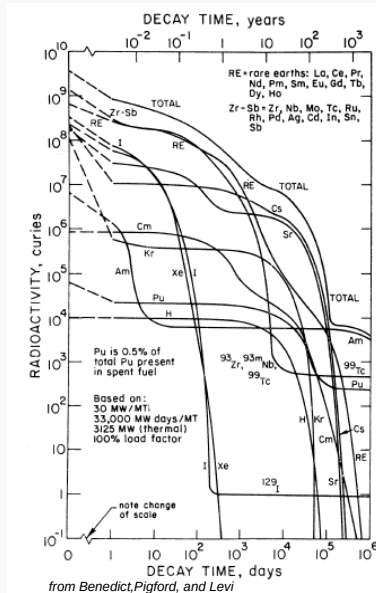
RADIOACTIVITY FROM PRINCIPAL FISSION PRODUCTS IN
IRRADIATED FUEL*

Nuclide	Half-Life	Activity at Discharge (1.0×10^6 Ci/yr)
³ H	12.3 yr	1.92×10^{-2}
⁸⁵ Kr	10.8 yr	0.308
⁹⁰ Sr	27.7 yr	2.11
⁹⁵ Zr	65.5 days	37.3
¹⁰⁶ Ru	368 days	14.8
¹²⁵ Sb	2.7 yr	0.237
¹²⁹ I	1.7×10^7 yr	1.01×10^{-6}
¹³¹ I	8.05 days	23.5
¹³³ Xe	5.27 days	43.9
¹³⁴ Cs	2.05 yr	6.7
¹³⁷ Cs	30.0 yr	2.94
¹⁴⁴ Ce	284 days	30.2
¹⁴⁷ Pm	4.4 yr	2.78
¹⁵¹ Sm	87 yr	3.41×10^{-2}
¹⁵⁴ Eu	16 yr	0.191
¹⁵⁵ Eu	1.81 yr	0.204

Source: Reprinted with permission from M. Benedict, T. Pigford, and H. Levi, *Nuclear Chemical Engineering*, 2nd ed., p. 354, McGraw-Hill Book Co., New York (1981).

*Fuel burnup is 33,000 MWD/MTU in a 1000-MW(e) PWR.

Spent Fuel Radioactivity Decay



Principal Actinides

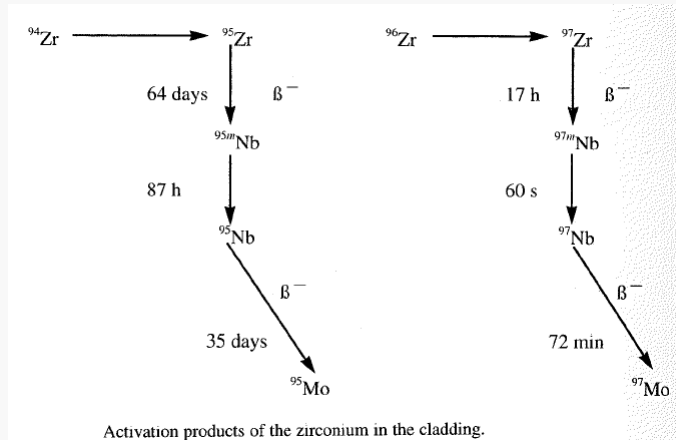
RADIOACTIVITY FROM PRINCIPAL ACTINIDES IN IRRADIATED FUEL*

Nuclide	Half-Life	Ci/yr	kg/MTU
²³⁴ U	2.47×10^5 yr	19.4	0.12
²³⁶ U	2.39×10^7 yr	7.22	4.18
²³⁷ Np	2.14×10^6 yr	14.4	0.75
²³⁸ Pu	2.85 yr	134	9.2×10^{-6}
²³⁹ Pu	86 yr	1.01×10^5	0.22
²⁴⁰ Pu	24,400 yr	8.82×10^3	5.28
²⁴¹ Pu	6,580 yr	1.30×10^4	2.17
²⁴² Pu	13.2 yr	2.81×10^6	1.02
²⁴² Pu	3.79×10^5 yr	37.6	0.35
²⁴¹ Am	458 yr	4.53×10^3	0.05
²⁴³ Am	7,950 yr	477	0.09
²⁴² Cm	163 days	4.40×10^5	4.9×10^{-3}
²⁴⁴ Cm	17.6 yr	7.38×10^4	3.3×10^{-2}

Source: Reprinted with permission from M. Benedict, T. Pigford, and H. Levi, *Nuclear Chemical Engineering*, 2nd ed., McGraw-Hill Book Co., New York (1981).

*Fuel burnup is 33,000 MWD/MTU in a 1000-MW(e) PWR, 150 days after discharge.

Cladding Activation



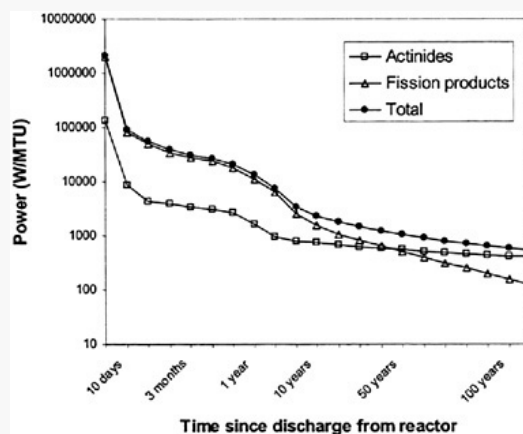
$^{93}\text{Nb} + \text{neutron} \rightarrow ^{94}\text{Nb}$ (20300 years half life) \

emits 2 gamma - rays; 0.702 and 0.871 MeV

Picture Courtesy:

R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management*, 2nd Edition, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999

Heat Generation in Spent Fuel



Source: Benedict, Pigford, and Levi

Irradiated Fuel Measurements

- What do we want to determine?
 - assembly serial number
 - fuel burnup
 - plutonium inventory of fuel
 - fuel assembly integrity
- What can we measure?
 - neutron emission [spontaneous fission and (α, n)]
 - gamma-rays
 - Cerenkov radiation

Non Destructive Burnup Measurements

- Cerenkov Radiation
- Single Fission Product Gamma-ray Activity

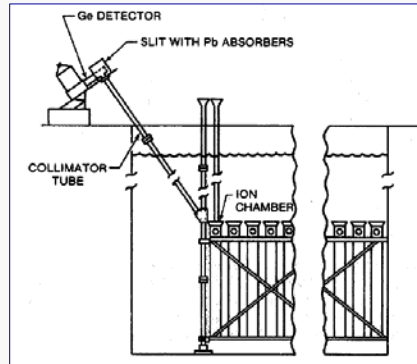
$$C' = \varepsilon P_B S_{atten} N_{fp} \lambda e^{-\lambda t_d}$$

$$BU(\text{atom}\%) = 100 \frac{N_{fp}}{Y_{fp} \cdot N_U}$$

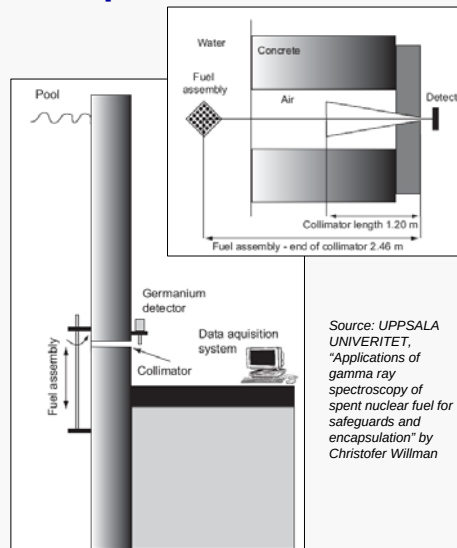
- Total Gamma-ray Activity
- Fission Product Activity Ratios
- Total Neutron Output

$$\text{neutron rate} = \alpha \cdot BU^\beta$$

Gamma-ray Measurement of Spent Fuel in Pool



Source: Passive Nondestructive Assay of Nuclear Materials NUREG/CR-5550 LA-UR-90-732 (PANDA Manual)



Source: UPPSALA UNIVERITET, "Applications of gamma ray spectroscopy of spent nuclear fuel for safeguards and encapsulation" by Christofer Willman

The mechanical installation used for performing the measurements. The figure shows the pool and pool wall from the side. The fuel assembly can be moved up and down as well as rotated around its vertical axis

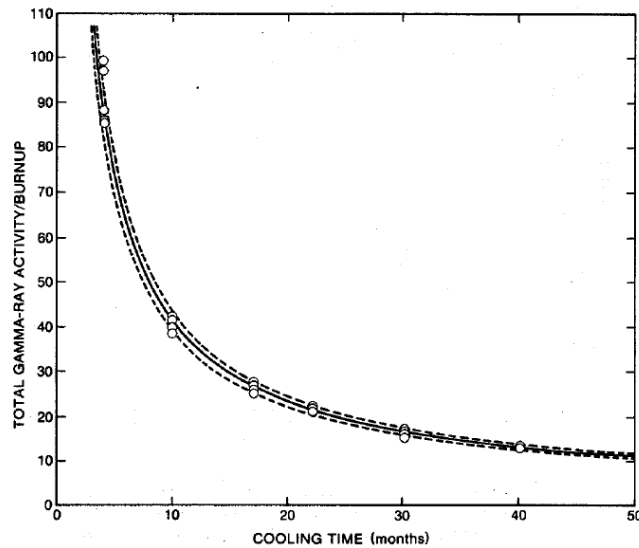
Atom Density and Gamma Activity of PWR Spent Fuel at 46 GWd/MTU

Fuel Exposure (GWd/tU)	¹³⁷ Cs		²³⁵ U		²³⁹ Pu	
	Atom Density ^a	661.6 keV (gammas/g U)	Atom Density ^a	185.72 keV (gammas/g U)	Atom Density ^a	413.69 keV (gammas/g U)
0	0.0	0.0	6.28 E+19	1.06 E+03	0.0	0.0
1.2	1.94 E+17	1.20 E+08	5.94 E+19	1.00 E+03	1.66 E+18	2.28 E+01
4.7	7.63 E+17	4.73 E+08	5.05 E+19	8.51 E+02	5.39 E+18	7.41 E+01
9.9	1.60 E+18	9.94 E+08	3.98 E+19	6.70 E+02	8.74 E+18	1.20 E+02
15.1	2.43 E+18	1.51 E+09	3.12 E+19	5.25 E+02	1.05 E+19	1.44 E+02
20.0	3.17 E+18	1.97 E+09	2.47 E+19	4.16 E+02	1.14 E+19	1.57 E+02
25.6	4.05 E+18	2.51 E+09	1.87 E+19	3.14 E+02	1.19 E+19	1.64 E+02
29.6	4.64 E+18	2.87 E+09	1.52 E+19	2.56 E+02	1.20 E+19	1.65 E+02
34.7	5.40 E+18	3.35 E+09	1.15 E+19	1.94 E+02	1.20 E+19	1.65 E+02
40.0	6.15 E+18	3.81 E+09	8.59 E+18	1.45 E+02	1.20 E+19	1.65 E+02
46.8	7.14 E+18	4.43 E+09	5.80 E+18	9.78 E+01	1.18 E+19	1.62 E+02

^aNumber of atoms per gram of initial heavy metal.

Source: Passive Nondestructive Assay of Nuclear Materials NUREG/CR-5550 LA-UR-90-732 (PANDA Manual)

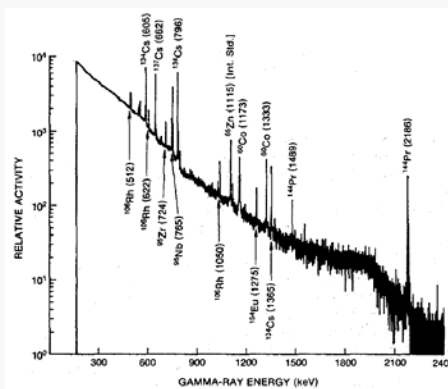
Total Gamma Activity Divided by Burnup Versus Cooling Time



$$\frac{C'}{BU} = f(t_d)$$

Source: Passive Nondestructive
Assay of Nuclear Materials
NUREG/CR-5550 LA-UR-90-732
(PANDA Manual)

Gamma Spectrum from 32 GWd/MTU PWR Fuel at 9 Months Cooling



Source: Passive Nondestructive Assay of Nuclear Materials NUREG/CR-5550 LA-UR-90-732 (PANDA Manual)

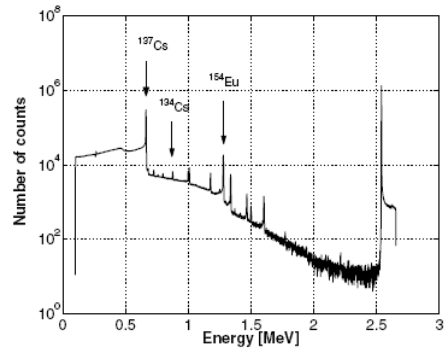
Measurable Gammas in PWR Spent Fuel

Fission Product Isotopes	Half-Life	Fission Yield in ²³⁵ U (%)	Fission Yield in ²³⁹ Pu (%)	Gamma-Ray Energy (keV)	Branching Ratio (%)
⁹⁵ Zr	64.0 days	6.50	4.89	724.2	43.1
⁹⁵ Nb	35.0 days	6.50	4.89	756.7	54.6
¹⁰³ Ru	39.4 days	3.04	6.95	765.8	99.8
¹⁰⁶ Ru-Rh	366.4 days	0.40		497.1	86.4
				610.3	5.4
			4.28	622.2	9.8
¹³⁴ Cs	2.06 yr	1.27 × 10 ⁻⁵ a	9.89 × 10 ⁻⁴ a	1050.5	1.6
				604.7	97.6
				795.8	85.4
				801.8	8.7
				1167.9	1.8
¹³⁷ Cs	30.17 yr	6.22	6.69	1365.1	3.0
¹⁴⁴ Ce-Pr	284.5 days	5.48	3.74	661.6	85.1
				696.5	1.3
				1489.2	0.3
				2185.6	0.7
¹⁵⁴ Eu	8.5 yr	2.69 × 10 ⁻⁶ a	9.22 × 10 ⁻⁵ a	996.3	10.3
				1004.8	17.4
				1274.4	35.5
Activation Products					
⁵⁴ Mn	312.2 days			834.8	100.0
⁶⁰ Co	70.3 days			811.1	99.0
	5.27 yr			1173.2	100.0
				1332.5	100.0

Three Important Gamma Emitting Fission Products

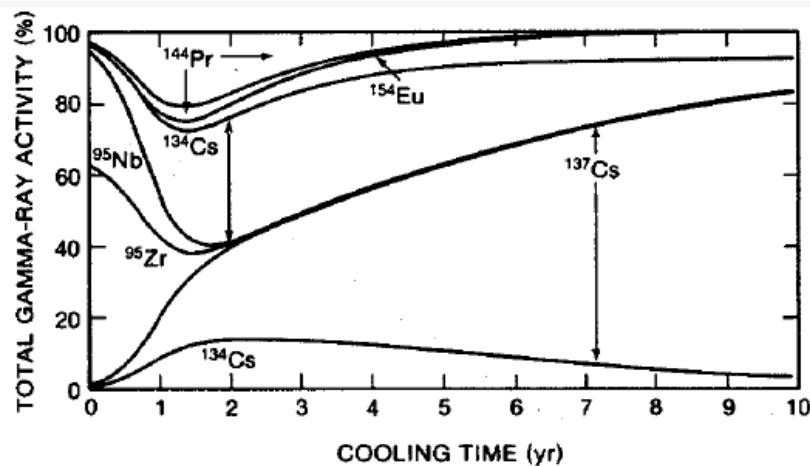
Isotope	Gamma-ray energy [keV]	Half life [years]
^{134}Cs	796, 1365	2.1
^{137}Cs	662	30.1
^{154}Eu	1275	8.6

Source: Passive Nondestructive Assay of Nuclear Materials NUREG/CR-5550 LA-UR-90-732 (PANDA Manual)



A typical gamma spectrum from a PWR spent fuel assembly at 47 GWd/tU and with 12 years cooling time

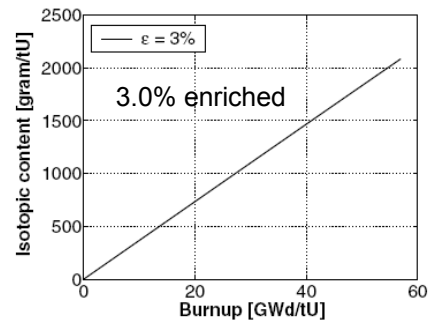
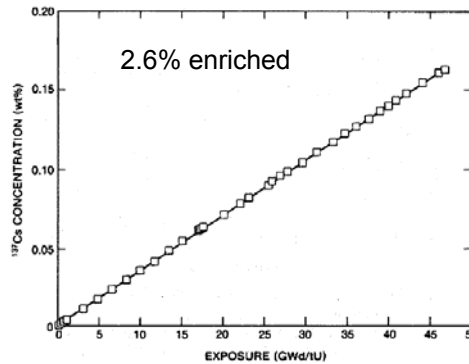
Percentage of Total Gamma-Ray Activity from Several Isotopes



Source: Passive Nondestructive Assay of Nuclear Materials NUREG/CR-5550 LA-UR-90-732 (PANDA Manual)

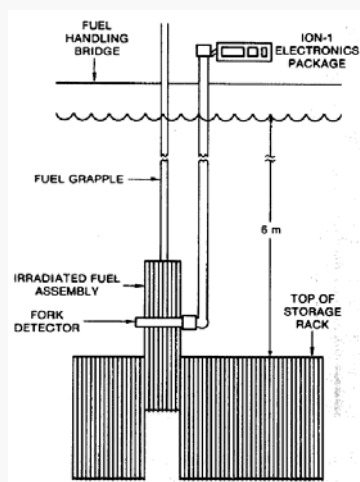
Burnup measurement methods

- Cesium 137 (an example relation given below)
- Cesium 134
- Europium 154
- Neutron production
- Fission product ratios

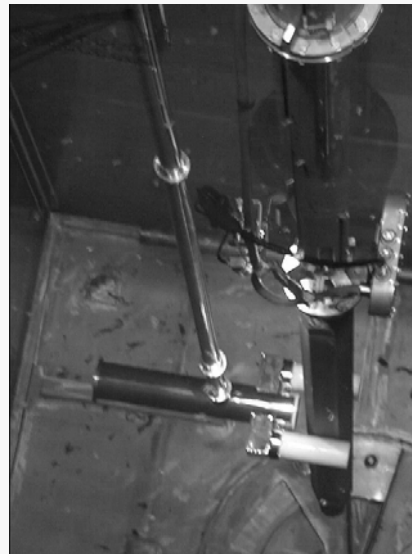


Source: Passive Nondestructive Assay of Nuclear Materials NUREG/CR-5550 LA-UR-90-732 (PANDA Manual)

Fork Detector: Combine n and γ Measurement



Source: Passive Nondestructive Assay of Nuclear Materials NUREG/CR-5550 LA-UR-90-732 (PANDA Manual)



References

1. International Atomic Energy Agency, "Spent fuel performance assessment and research," IAEA-TECDOC-1343, Spent Fuel Performance Assessment and Research (SPAR) 1997–2001, Vienna, 2003
2. I.L. Rybalchenko and J.P. Colton, "Spent Fuel Management," IAEA Bulletin - Vol.23, No. 2, IAEA, Vienna, 1981
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4. International Atomic Energy Agency, "Fuel failure in water reactors: Causes and mitigation," IAEA-TECDOC-1345, Vienna, March 2003
5. F. von Hippel, "Interim Dry-cask Storage vs. Spent Fuel Reprocessing," Co-chair, International Panel on Fissile Material (IPFM), Senate Staff Briefing, July 2009
http://www.armscontrolcenter.org/assets/ppt/vonhippel_PP.ppt
6. World Nuclear Association, "Processing of Used Nuclear Fuel," <http://www.world-nuclear.org/>
7. J. Cerami, "Nuclear Fuel Reprocessing: Now We're Ready(?)," Presentation Slides, <http://www.kentlaw.edu/faculty/fbosselman/classes/EnergyLawSp07/PowerPoints/CeramiNuclearFuelReprocessing.ppt>
8. W. Charlton and D. Boyle, Course Presentation Slides, Private Communication

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- Readings for the lectures and selected problem sets will be assigned from this text

Reading Assignments

- C&T Ch. 7, 8, 9
- Assorted papers

Exercises

- C&T problems 6.2, 6.6
- Develop a position paper associated with storing spent fuel vs pursuing a reprocessing strategy (consider both an indigenous or external reprocessing capability). What aspects drive the costs? What political ramifications are there? Legal aspects?

End of Lecture 8

Spent Fuel Recycling Lecture 9

GNEII Pilot Course

February – May 2011

Khalifa University Science, Technology and Research,

United Arab Emirates, Abu Dhabi

(William Charlton)

Texas A&M University, USA

SAND 2010_____

Sandia National Laboratories is a multi-program laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

Spent Fuel Recycling

- **Main Teaching Points**

1. Purpose of Spent Fuel Recycling
2. Prominent Spent Fuel Recycling Methods

Spent Fuel Recycling

- **Key Terms** – reprocessing, PUREX, UREX
- **Key Concepts** – objectives of spent fuel recycling, steps of reprocessing process, necessity for interim storage before recycling, nuclear material proliferation concern of spent fuel reprocessing

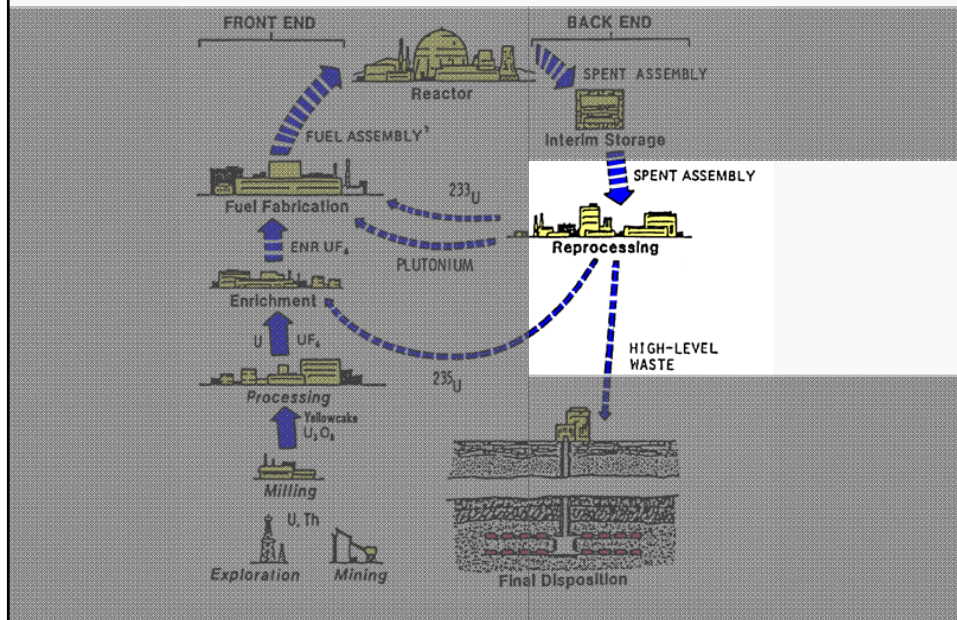
Spent Fuel Recycling

- **Desired Student Outcomes:**
 1. Understand why spent fuel recycling is needed
 2. Be familiar with important spent fuel reprocessing methods

Assumptions and Prerequisites

- Successful completion of GNEII weeks one and two
- Bachelor's degree
- College level math

Nuclear Fuel Cycle



Spent Fuel Recycling: Essential Feature in Closed Fuel Cycle

- Spent fuel contains most of the fertile material (^{238}U), only ^{235}U is depleted
- Spent fuel contains appreciable concentrations of valuable fissile nuclides (^{235}U and plutonium)
- Also, contains large amounts of radioactivity, neutron absorbing fission products
- Principal objectives of spent fuel **recycling or reprocessing** are;
 - To recover uranium and plutonium for reuse as nuclear fuel
 - To remove radioactive and neutron-absorbing fission products
 - To convert the radioactive constituents of spent fuel into forms suitable for safe, long-term storage
 - Fission products ^{90}Sr and ^{137}Cs recovered can be used as radiation sources in various applications

Interim Storage of Spent Fuel Before Recycling

- Nominal design period for storing spent fuel before reprocessing (recycling) is 150 days
- 150 days of spent fuel cooling in wet storage allows the radioactivity and decay power to decrease for allowing reprocessing operations
- Other reasons for 150 days cooling are;
 - All gaseous fission products except tritium, ^{85}Kr , and ^{129}I decay to inconsequential levels during this cooling period
 - To reduce ^{131}I activity (8 day half life) to manageable levels
 - To complete decay of 2.35-day half life ^{239}Np to ^{239}Pu
- Some reprocessing plants design is such that the spent fuel cooling might be much longer than 150 days

Spent Fuel Recycling Process - 1

- The most common method used for spent fuel reprocessing is PUREX (Plutonium URanium Extraction)
- PUREX is a chemical process used to separate uranium, plutonium and radioactive fission products into three streams
- Reprocessed uranium and plutonium can be mixed to make mixed oxide fuel ($\text{UO}_2\text{-PuO}_2$) for loading back into PWRs
- PUREX is a chemical solvent extraction process which uses chemical like tributyl phosphate (TBP) and a hydrocarbon diluent to extract uranyl nitrate and plutonium nitrate from an aqueous solution containing nitric acid

Spent Fuel Recycling Process - 2

- Principal steps of PUREX
 - Cooled spent fuel is chopped into small pieces by mechanical shearing or sawing
 - Fuel and cladding are charged to a dissolver and reacted with nitric acid
 - Cladding hulls are separated, and washed with water (to remove most of the fission products and fuel sticking to cladding), and is packaged, and shipped to waste storage
 - Using the solvent extraction chemical process with TBP 99% of fission products are separated out from plutonium + uranium solution
 - By appropriate chemical methods plutonium and uranium are separated for new MOX fuel fabrication

Spent Fuel Recycling Process – Other Method

- There is nuclear material proliferation concern with respect to fuel reprocessing using PUREX
- So, methods like UREX are being investigated where in uranium and plutonium are not separated but are co-extracted

References

1. International Atomic Energy Agency, "Spent fuel performance assessment and research," IAEA-TECDOC-1343, Spent Fuel Performance Assessment and Research (SPAR) 1997–2001, Vienna, 2003
2. I.L. Rybalchenko and J.P. Colton, "Spent Fuel Management," IAEA Bulletin - Vol.23, No. 2, IAEA, Vienna, 1981
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End of Lecture 9

Week 3 - Day 4

Nuclear Fuel Cycle

Lecture 10 – Mixed Oxide (MOX) Fuel

Lecture 11 – Radioactive Waste Management

Mixed Oxide (MOX) Fuel

Lecture 10

GNEII Pilot Course

February – May 2011

Khalifa University Science, Technology and Research,

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Mixed Oxide (MOX) Fuel

- **Main Teaching Points**
 1. MOX Fuel characteristics and use
 2. MOX experience
 3. Advantages of recycling spent nuclear fuel

Mixed Oxide (MOX) Fuel

- **Key Terms** – MOX, Recycling,
- **Key Concepts** – MOX and UO₂ irradiation differences, need for recycling spent nuclear fuel, reduction in repository volume due to recycling

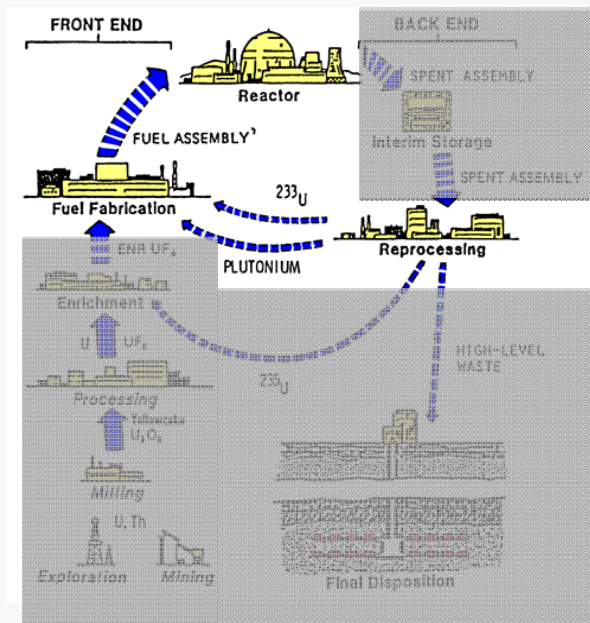
Mixed Oxide (MOX) Fuel

- **Desired Student Outcomes:**
 1. Utility of using mixed oxide fuel
 2. Be able to describe what makes up spent fuel and which portion is waste
 3. Difference between UO_2 and MOX irradiation

Assumptions and Prerequisites

- Successful completion of GNEII weeks one and two
- Bachelor's degree
- College level math

Nuclear Fuel Cycle



MOX Highlights (1 of 2)

1. What fraction of used fuel discharged from a typical LWR is plutonium? **About 1%**
~ 50% Pu-239
~ 15% Pu-241
2. How much plutonium is discharged each year from worldwide nuclear power reactors? **About 70 M-Tonnes**

Inventory of Separated Recyclable Materials
Source: NEA 2007.

	Quantity - tonnes	Natural U equivalent - tonnes
Plutonium from reprocessed fuel	320	60,000
Uranium from reprocessed fuel	45,000	50,000
Ex-military plutonium	70	15,000
Ex-military high-enriched uranium	230	70,000

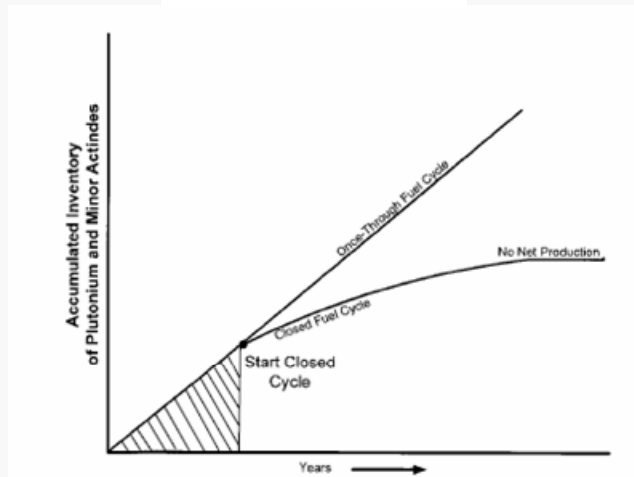
MOX Highlights (2 of 2)

- **MOX = Physical mixture of UO_2 and PuO_2**
 - ~ 7-9% reactor plutonium mixed with depleted U up to ~ 4.35% U-235
 - Use of weapons plutonium allows ~ 5% plutonium in the mix
- **What are “time constraints” for fabricating MOX from reprocessed fuel?**
- **Am-241 and Pu-238 build up**
 - Reactor plutonium over 5 years old requires “specialized” MOX plant
- **Why are Fast Reactors “needed” for multiple recycling of plutonium?**
- **All transuranic isotopes are fissionable**
 - Isotopic degradation limits plutonium recycle potential in LWRs
 - Most spent MOX fuel is currently stored awaiting deployment of fast reactors
 - MOX fuel at 45 GWd/tU:

37% Pu-239	32% Pu-240	16% Pu-241
12% Pu-242	4% Pu-238	

Generic Accumulation of Plutonium and Minor Actinides (Np, Am, Cm)

Why MOX is Important

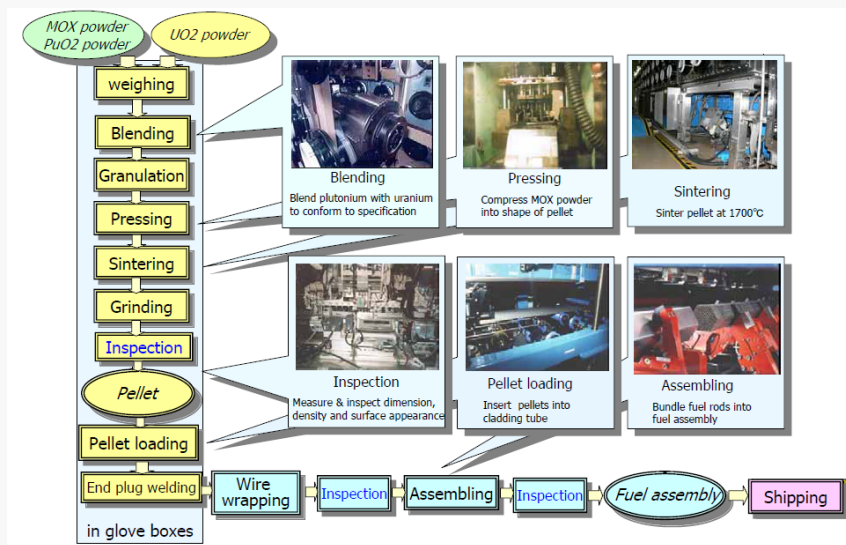


Recycle MOX to Reduce Load on Spent Fuel Repository

- Main Source of Decay Heat = Pu-241-Am, Pu-238, Cm-244, Cs-137, Sr-90, and their decay products
- Removal of Am and Pu increases repository capacity 5 times
- Removing Cs and Sr along with Am and Pu increases capacity 42 times!

OBJECTIVE: "Partition" actinides from LWR spent fuel and subsequently "transmute" them in FBRs (or LWRs?)

How MOX Fuel Assemblies are made?



Source: http://insaf-net.org/mox_wg/JAEA.pdf

MOX Fuel from Dismantled Nuclear Weapons

- Apart from obtaining plutonium from spent fuel using reprocessing to prepare MOX fuel, plutonium can be obtained from dismantle nuclear explosive devices

MOX Fuel is Widely used Today

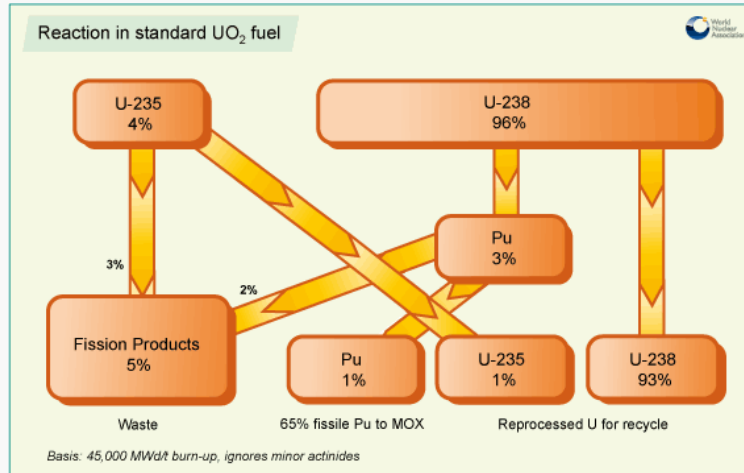
Europe

- 40 reactors licensed to use MOX
- 30 are using MOX (Belgium, Switzerland, Germany, France)

Japan

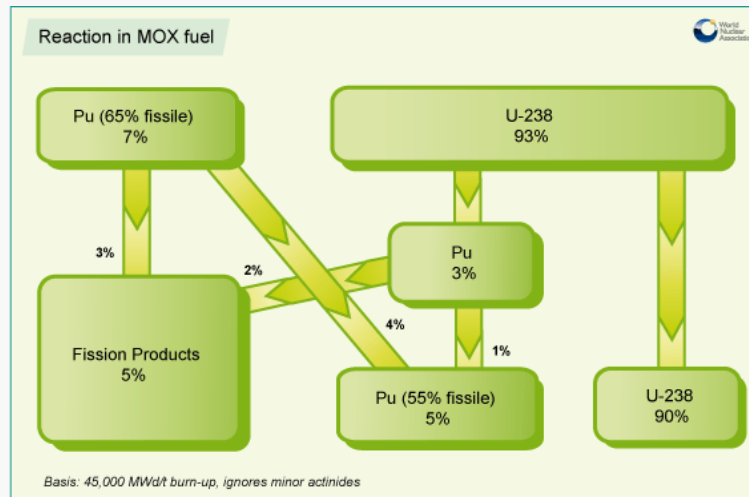
- Plans to use MOX in up to 20 reactors
- Typically one-third of core is MOX
- 50% MOX assemblies in some current reactors
- Advanced LWRs (EPR or AP-1000) will take full MOX loads

Differences between irradiation of standard UO_2 and MOX fuel in reactors - 1



Source: World Nuclear Association
<http://www.world-nuclear.org/info/inf29.html>

Differences between irradiation of standard UO_2 and MOX fuel in reactors - 2



Source: World Nuclear Association
<http://www.world-nuclear.org/info/inf29.html>

References

1. International Atomic Energy Agency, "Spent fuel performance assessment and research," IAEA-TECREP-415, TECHNICAL REPORTS SERIES No. 415, Vienna, 2003
2. World Nuclear Association, "Radioactive Waste," <http://www.world-nuclear.org/>
3. World Nuclear Association, "Mixed Oxide (MOX) Fuel," <http://www.world-nuclear.org/>
4. World Nuclear Association, "Japanese Waste and MOX Shipments From Europe," <http://www.world-nuclear.org/>
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Reading Assignments

- C&T review
- Assorted papers

Exercises

- C&T problems 7.3, 7.5, example 7.4
- Identify possible storage options for spent nuclear fuel

End of Lecture 10

Week 3 - Day 4

Nuclear Fuel Cycle

Lecture 10 – Mixed Oxide (MOX) Fuel

Lecture 11 – Radioactive Waste Management

Radioactive Waste Management

Lecture 11

GNEII Pilot Course

February – May 2011

KUSTAR, Abu Dhabi

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Radioactive Waste Management

- **Main Teaching Points**

1. Waste storage options
2. Waste characteristics
3. Types of waste
4. Waste Classification
5. Waste Management

Radioactive Waste Management

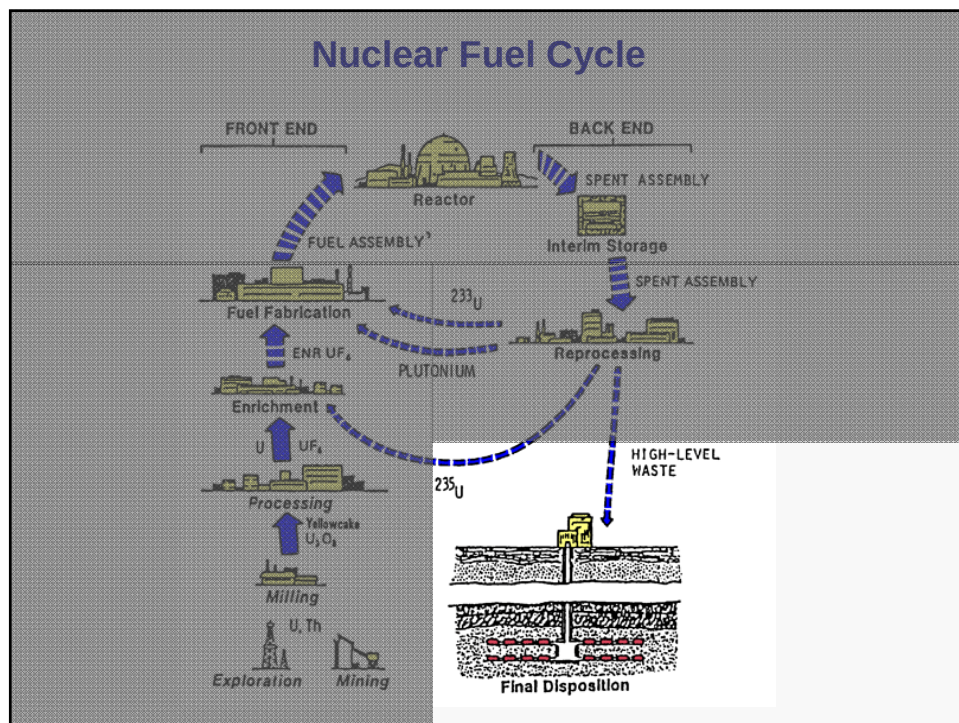
- **Key Terms** – transuranic waste, low and high level waste, activity, half-life, storage tanks, geological repository, vitrification, bituminization
- **Key Concepts** – parts of spent fuel that cannot be reused, methods used for actinide separation and disposal, classification of low and high level waste, principles of waste management, waste management planning, classification of radioactive wastes, methods of high level waste management

Radioactive Waste Management

- **Desired Student Outcomes:**
 1. Be able to describe what makes up spent fuel and which portion is waste
 2. Understand how the various storage methods remove waste from the biosphere
 3. Know the difference between low-level and high-level waste
 4. Be familiar with the categories of radioactive waste
 5. Understand the principles of radioactive waste management
 6. Understand about waste management planning
 7. Knowledge about HLW management

Assumptions and Prerequisites

- Successful completion of GNEII weeks one and two
- Bachelor's degree
- College level math



Introduction

- Safe management of radioactive waste forms an integral part of the nuclear fuel cycle
- Waste management has prime importance to remove the apprehensions in the public mind
- Safe management of radio-active wastes should encompass collection, processing, storage and proper disposal
- **Storage** – interim storage of high level radioactive waste under surveillance to assure public and environmental safety and will be retrieved to process after 50 to 100 years
- **Disposal** – means no intention to retrieve and stops all controls on surveillance and safety is assured by methods other than surveillance

Classification of Radioactive Wastes

- Radioactive wastes can be classified on the basis of a number of considerations
- They can be classified as solid, liquid, and gaseous waste based on their physical form
- They can be classified as low-level, intermediate-level, and high-level based on activity content
- They can be classified on the basis of 'what goes where' i.e. on the disposal method adopted
- They can be classified based on their source of origin in the different phase of the nuclear fuel cycle: mining and milling waste, fuel fabrication waste, reactor wastes and wastes from spent fuel reprocessing

High Level Waste (HLW)

1. **Spent Fuel:** irradiated commercial reactor fuel.
2. **Reprocessing Waste:** liquid waste from solvent extraction cycles in reprocessing. Also the solids into which liquid wastes may have been converted.

Transuranic Wastes (TRU)

- Wastes containing alpha-emitting isotopes with atomic number >92 and concentrations greater than 100 nCi per gram of waste
- Another definition is radioactive waste that is not classified as HLW but contains an activity of more than 100 nCi/g from alpha-emitting TRU isotopes with half-lives greater than 20 years;

Categorization of radioactive wastes based on disposal method recommended by IAEA

Category	Disposal method	Examples
High beta-gamma, significant alpha, high heat output	Deep geological formations in salt or granite medium	Vitrified glass wastes arising in spent fuel reprocessing, conditioned spent fuel
Intermediate beta-gamma, significant alpha, low heat output	Deep geological formations as above	Spent fuel cladding hulls, intermediate level reprocessing wastes solidified with bitumen, cement or synthetic polymer
Low beta-gamma, significant long-lived, low heat output	As above	Wastes from plutonium (MOX) fuel fabrication plants
Low beta-gamma, significant long-lived, low heat output	Shallow land burial with or without engineered barriers	Reactor wastes

Significant Radio-nuclides in Wastes from Nuclear Fuel Cycle

Source	Disposal method
<u>Gaseous wastes – fuel cycle</u>	
Uranium mill tailings	^{222}Rn
Light water reactors	Short lived fission gases, ^{138}Xe , ^{88}Kr , ^{135}Xe , ^{87}Kr , ^{133}Xe
Spent fuel reprocessing off gas	^{14}C , ^{129}I , ^{131}I , ^{85}Kr
<u>Liquid/Solid wastes – fuel cycle</u>	Deep geological formations as above
Uranium mining and milling	^{226}Ra
Recycle fuel fabrication	^{238}Pu , ^{239}Pu , ^{240}Pu , ^{241}Am
Reactor wastes	^{60}Co , ^{90}Sr , ^{137}Cs , ^{131}I
Spent fuel reprocessing waste	^{90}Sr , ^{137}Cs , ^{106}Ru , actinides

Waste Management Planning

- Establish procedures for minimization, proper segregation, collection, interim storage and transport of radioactive wastes
- Development of processes for treatment and conditioning for conversion to acceptable waste forms for interim storage or disposal and reduction of volume
- Define and implement acceptable waste disposal concepts
- Specific investigations and surveillance programs for disposal sites

Desirable Characteristics of Waste Forms

- Solid form, preferably a rigid monolith of low surface area
- Low leach rates for radio-nuclides of interest in water
- Good thermal, chemical, mechanical and radiation stability
- Should be compatible with the surrounding medium

General Principles of Waste Management

- Three basic concepts applied to radioactive waste management are;
 - Delay and decay
 - Dilute and disperse
 - Concentrate and contain
- First concept is applicable to short-lived wastes
- Second concept is applicable to
 - Large volumes of low-level liquid effluents from nuclear installations located on coastal or near large water bodies
 - Atmospheric discharge of gaseous waste through tall stacks and subsequent dispersion
- Third concept is the one applied to waste which cannot be managed on the basis of first two

Concentration and Conditioning of Wastes

Waste	Process
Compressible solid waste	Baling
Combustible solid waste	Incineration, fix fly ash in suitable medium
Liquid effluents	<u>Evaporation</u> Discharge condensate as low-level effluent; fix concentrate in suitable medium as solid waste <u>Co-precipitation</u> After adjusting pH etc. add suitable carriers to precipitate radio-nuclides; filter slurry; sludge to be fixed in suitable medium. Eg. Copper ferrocynide for Cs, Ba, sulfate for Sr <u>Ion-exchange</u> Synthetic organic ion-exchange materials will concentrate radio-nuclide from the effluent; on elution a concentrated re-generant waste stream will result which can be subjected to a co-precipitation process. Inorganic ion-exchanges such as vermiculate can be used to selectively retain certain radio-nuclides and can be disposed of as solid waste

Media commonly used for immobilizing chemical sludge and slurries are cement, bitumen and synthetic polymers

Practices Used to Treat and Immobilize Transuranic Wastes

Country	Treatment ^a			Immobilization		
	Decontamination	Incineration	Size reduction	Cement	Bitumen	Other
Belgium	—	X	—	X	X	X
China	—	—	—	—	—	—
France	P	X	X	X	X	X
West Germany	X	X	X	X	—	—
India	—	P	P	—	—	—
Italy	—	P	—	—	X	—
Japan	—	P	P	—	—	—
United Kingdom	X	X	P	X	—	—
United States	P	X	P	—	—	—
USSR	—	X	X	X	X	—

^a "X, Practice in use; P, practice proposed.

Practices Used to for Disposal of Transuranic Waste

Country	Facilities used in disposal of transuranic waste			
	Interim storage	Sea disposal	Geologic disposal	Deep-well disposal
Belgium	X	X	P	—
China	X	—	—	—
France	X	—	P	—
West Germany	X	—	P	—
India	X	—	P	—
Italy	X	—	P	—
Japan	X	—	P	—
United Kingdom	X	X	P	—
United States	X	—	X	X
USSR	X	—	P	X

^aX, Practice in use; P, practice proposed.

Sources of TRU Wastes

- Fuel reprocessing.
- Waste immobilization facilities.
- Fuel fabrication of recycled Pu (i.e., MOX).
- Weapons material production.
- Decommissioning of nuclear facilities.

Desirable Acceptance Criteria for Waste - 1

- Cannot have the following characteristics:
 - Gas generation
 - Combustibility
 - Dispersible
 - Explosive or compressed gases
 - Pyrophoric materials

Desirable Acceptance Criteria for Waste - 2

- **Things to Verify:**
 - Toxic or corrosive compounds
 - Proper Waste containers
 - Proper Waste package handling
 - Proper Waste package weight
 - Proper Waste package size
- **Things to Monitor:**
 - Surface dose rate
 - Surface contamination
 - Thermal power
 - Criticality
 - Documentation

Waste Package Certification

- **Multiple Test Methods**
 - Gamma spectroscopy
 - X-ray techniques including fluoroscopy
 - Passive computed tomography
 - Active computed tomography
 - Combinations of these
- **Newest approach is to use active and passive computed tomography**
 - Employs multiple detectors
 - May employ multiple sources

Treatment of HLW

- High-level wastes produced are usually stored in tanks – stainless steel tanks in concrete vault
- Radioactivity concentration can be in the range of 5000-10000 curies per liter (solution should be cooled)
- Storage in liquid form is only temporary arrangement to provide for operational flexibility
- Need to convert to a stable, solid form using a process known as vitrification

Vitrification of HLW

- In vitrification process calcine is formed at a temperature of 500-600 degree Celsius
- Water boils off
- Nitrates are converted to oxides
- Glass forming 'frit' is now added (mainly borax and silica) in the correct ratios and the temperature is raised to 1000 degree Celsius
- Upon cooling a glass like borosilicate glass product is formed
- The selected glasses have proved very resistant to the effect of heat and radiation
- These glass product from vitrification can be stored and disposed

References

1. International Atomic Energy Agency, "Spent fuel performance assessment and research," IAEA-TECREP-415, TECHNICAL REPORTS SERIES No. 415, Vienna, 2003
2. World Nuclear Association, "Radioactive Waste," <http://www.world-nuclear.org/>
3. World Nuclear Association, "Mixed Oxide (MOX) Fuel," <http://www.world-nuclear.org/>
4. World Nuclear Association, "Japanese Waste and MOX Shipments From Europe," <http://www.world-nuclear.org/>
5. W. Charlton and D. Boyle, Course Presentation Slides, Private Communication

List of Required Texts

- The following textbook is required:
 - R.G. Cochran and N. Tsoulfanidis, *The Nuclear Fuel Cycle: Analysis and Management, 2nd Edition*, ISBN: 0894484516, American Nuclear Society, La Grange Park, IL, 1999
- This text is available for purchase online at <http://www.ans.org/store/vi-350015>
- Readings for the lectures and selected problem sets will be assigned from this text

Reading Assignments

- C&T review
- Assorted papers

Exercises

- C&T problems 7.3, 7.5, example 7.4
- Identify possible storage options for spent nuclear fuel

End of Lecture 11