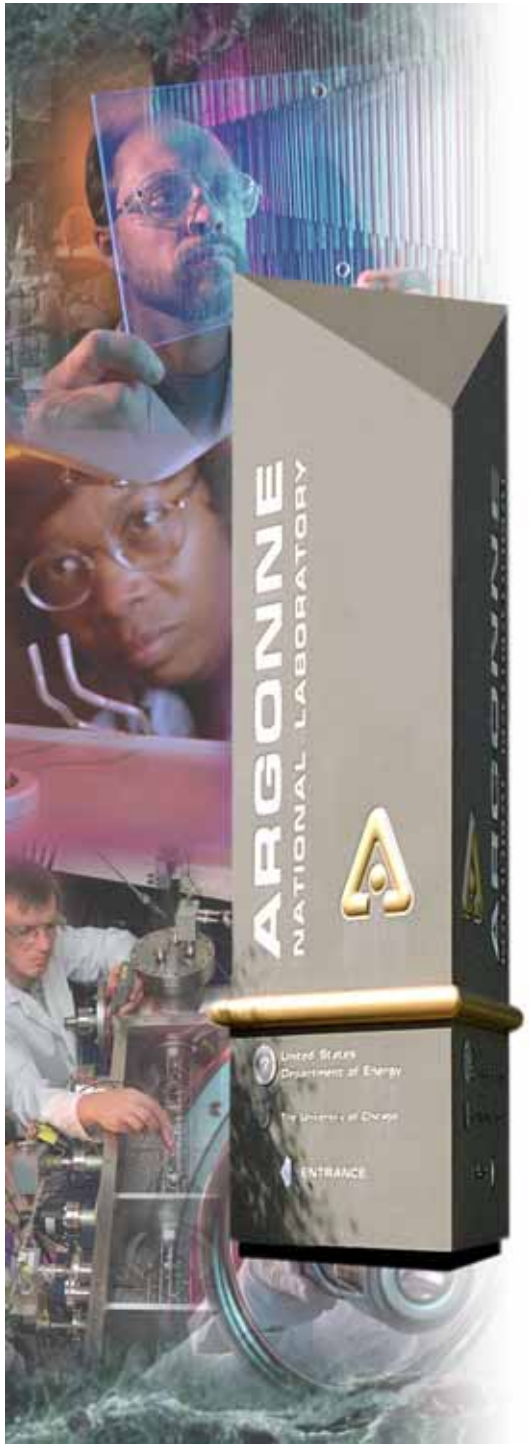

Systems Studies Strategies for Transitioning to Advanced Fuel Cycles and Reactor Systems

Workshop for Graduate Students from the Big 10 Nuclear Engineering Departments
July 22, 2004

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Nuclear Engineering Division

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Overview

- **Advanced Fuel Cycle Initiative (AFCI) Systems Analysis**
  - Waste transmutation objectives
  - Repository benefit and performance assessment
- **Generation-IV Systems Analysis**
- **Transition to Advanced Nuclear Fuel Cycle**
  - General implementation phases
  - Illustrative fuel cycle scenarios
  - Equilibrium end states

Advanced Fuel Cycle Systems Analysis

- **Systems analysis of the nuclear fuel cycle encompasses a wide variety of activities, three main areas pursued in AFCI Program**
  - Transmutation Options (ANL, ORNL, BNL, INEEL)
    - *Systematic evaluations of LWR recycle, LWR MOX options and low conversion fast reactors*
      - Assess feasibility and quantify performance
    - *Transmutation criteria*
  - Benefits (ANL, INEEL, LLNL, SNL, ORNL, SRS)
    - *Repository*
      - Quantification of capacity benefits based on heat load
    - *Economic – work just starting in FY03*
    - *Non-proliferation*
  - Global Studies (INEEL, ANL, LANL, SNL)
    - *Dynamic study of key scenarios*

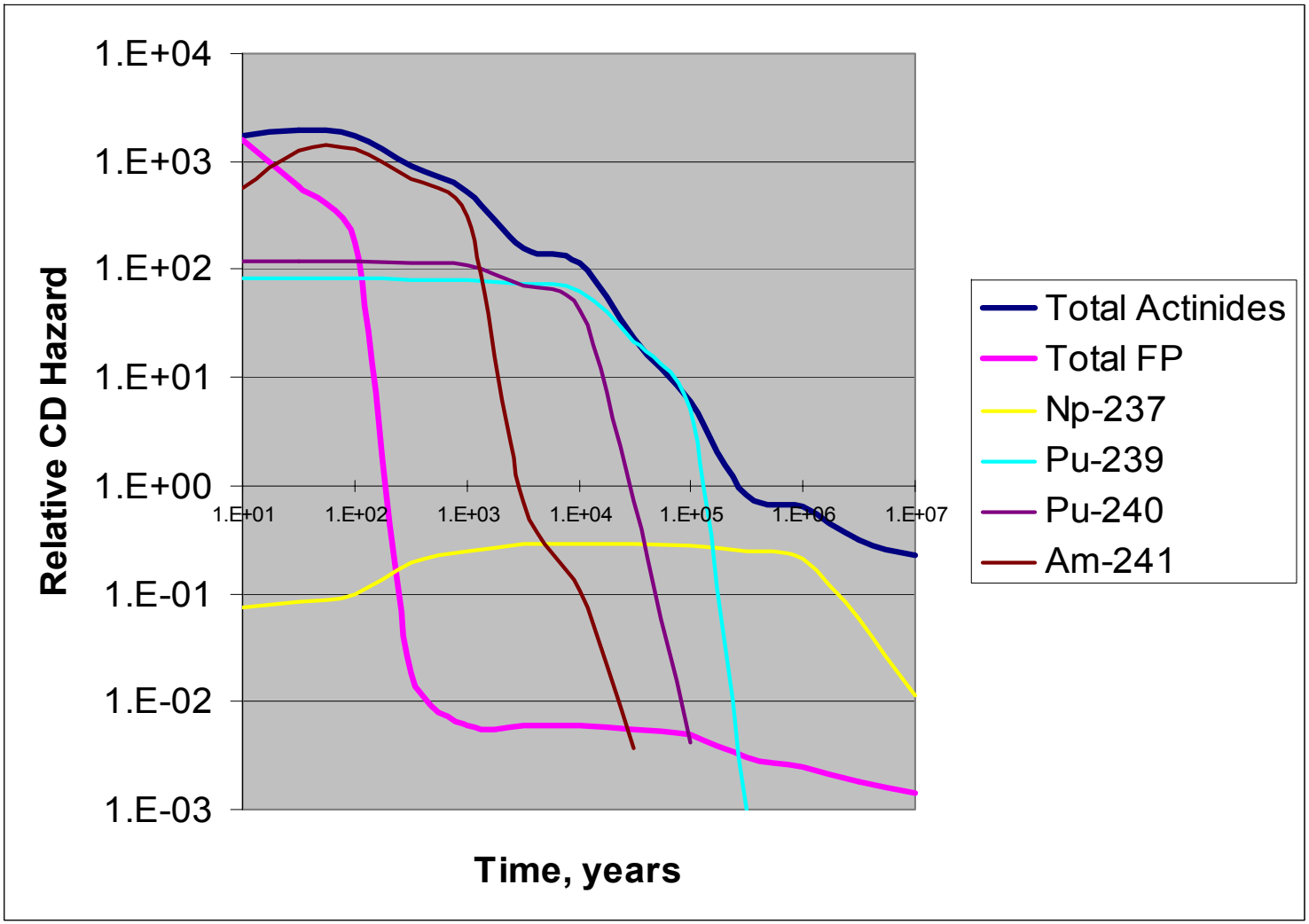
AFCI Waste Transmutation Goals

- **Benefit to the waste disposal system**
  - Focus is on Yucca Mountain geological repository
  - Increase capacity (minimize waste volume)
  - Reduction of long-term waste radiotoxicity
  - Improve performance (reduce released dose rates)
- **Reduce proliferation risk**
  - Prevent material diversion at all points in nuclear fuel cycle
  - Avoid fissile material discharge to repository (plutonium mine)
- **Improve prospects for nuclear power**
  - Provide improved waste solution (including economics)
  - Develop/implement advanced nuclear technologies

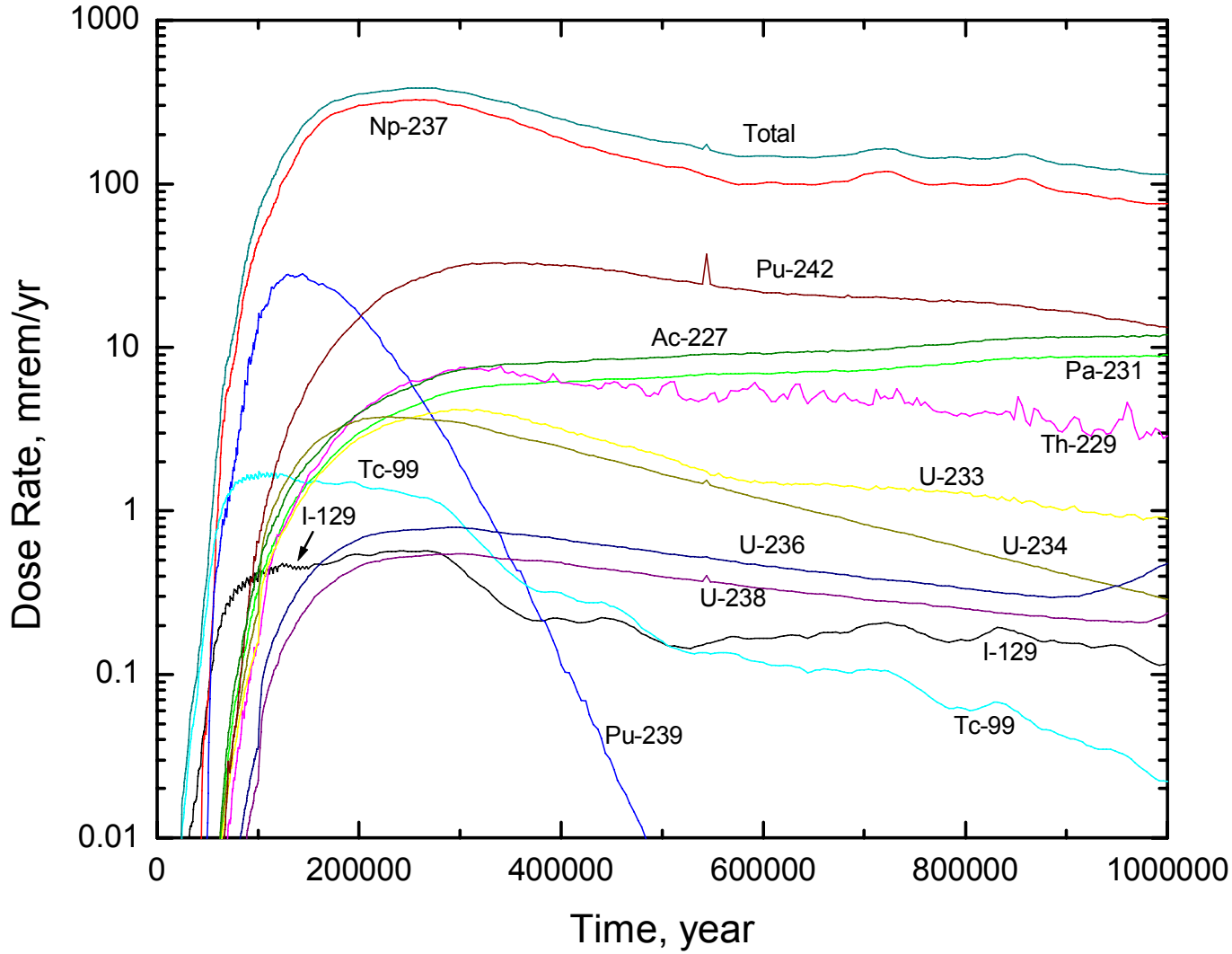
Environmental Hazard and Risk

- **Radiotoxicity quantifies the effect of exposure (<u>hazard</u>)**
  - Effectively assumes complete release and uptake
- **Fission products dominate in short term**
  - Fall well below the original toxicity after 200 years
- **Am-241 dominates in the 1,000 year time frame**
- **Pu-239 and Pu-240 dominate in the 10,000 to 100,000 time**
- **Np-237 dominates the long term (>10^5 years) hazard**
  - Comes primarily from Am-241 decay
  - Am-241 results from Pu-241 decay (and Cm-245 decay)
- **Repository environment will impact radiological <u>risk</u>**
  - All material is contained for ~10,000 years
  - Plutonium moves slowly, FP quickly through environment
  - Long-lived fission products (Tc-99 and I-129) dominate early
  - Maximum dose results from Np-237 in long-term

Radiotoxicity of LWR Spent Fuel



Yucca Mountain Estimated Dose Rates for Direct Disposal of LWR Spent Fuel



Yucca Mountain Performance Assessment Estimated Released Dose Rates

- **Dose criteria set limits on the peak dose rate during the regulatory period of 10,000 years**
  - limit of 15 mrem/year for the maximally reasonably exposed individual
  - long-term dose evaluated in performance assessment
  - However, EPA limit vacated in recent court ruling
- **Actinides (Np-237 in particular) dominate the peak dose rate**
  - Peak dose rate >100 mrem/yr in the 100,000 year time frame
  - Extends out to 1,000,000 years before the Np starts to decay
- **Contribution of LLFPs dominate in the short-term**
  - Tc-99 does peaks at 2 mrem/yr, and I-129 at 0.7 mrem/yr
  - Actinides do not escape for first 40,000 years
  - LLFP release is not solubility limited, and they are more mobile
  - Thus, *container design/failure* dictates LLFP dose rate

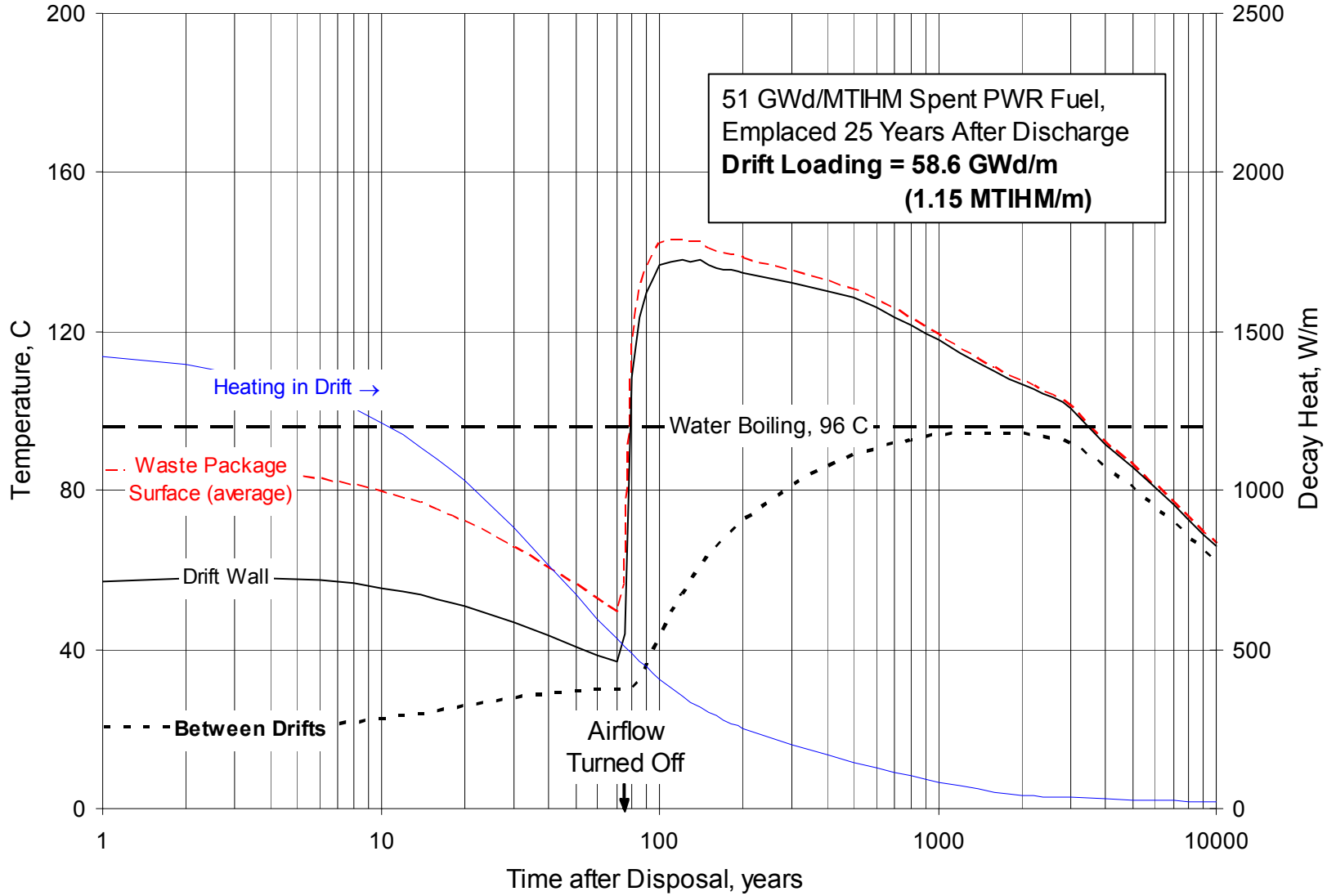
Repository Capacity and Loading

- **Spent fuel from current U.S. nuclear plants will exceed target Yucca Mountain capacity in 2025**
- **Goal is to provide advanced fuel cycle which allows continued nuclear production <u>and</u> avoids 2nd repository**
  - This is achieved by both reduced waste quantities and improved loading of the geologic repository
- **Thermal criteria set a number of temperature limits constraining the design and operation for the Yucca Mountain Repository; limits are still evolving, including**
  - peak temperature below the local boiling point (96 ºC) at all times midway between adjacent drifts
  - peak temperature of the drift wall below 200 ºC at all time
  - all criteria are met with the Yucca Mountain Repository design

Direct Disposal of Spent PWR Fuel

- **Spent PWR fuel is used as the reference spent fuel for direct disposal, with the following characteristics**
  - 51 GWD/MTIHM discharge burnup
  - placement in the repository 25 years after discharge
- **The response of the repository is modeled as follows**
  - a drift near the center of the repository is used as the highest temperatures would be expected in this region
    - *allows a small section of the existing 3-D model to be used*
  - forced ventilation for 75 years
- **Maximum drift loading is 1.15 MTIHM / m, representing material that produced 58.6 GWd of energy being stored per meter**
  - initial linear heat load is 1.45 kW / m
  - no margin is provided for uncertainties; used as reference point
    - *all other cases are evaluated in the same manner*

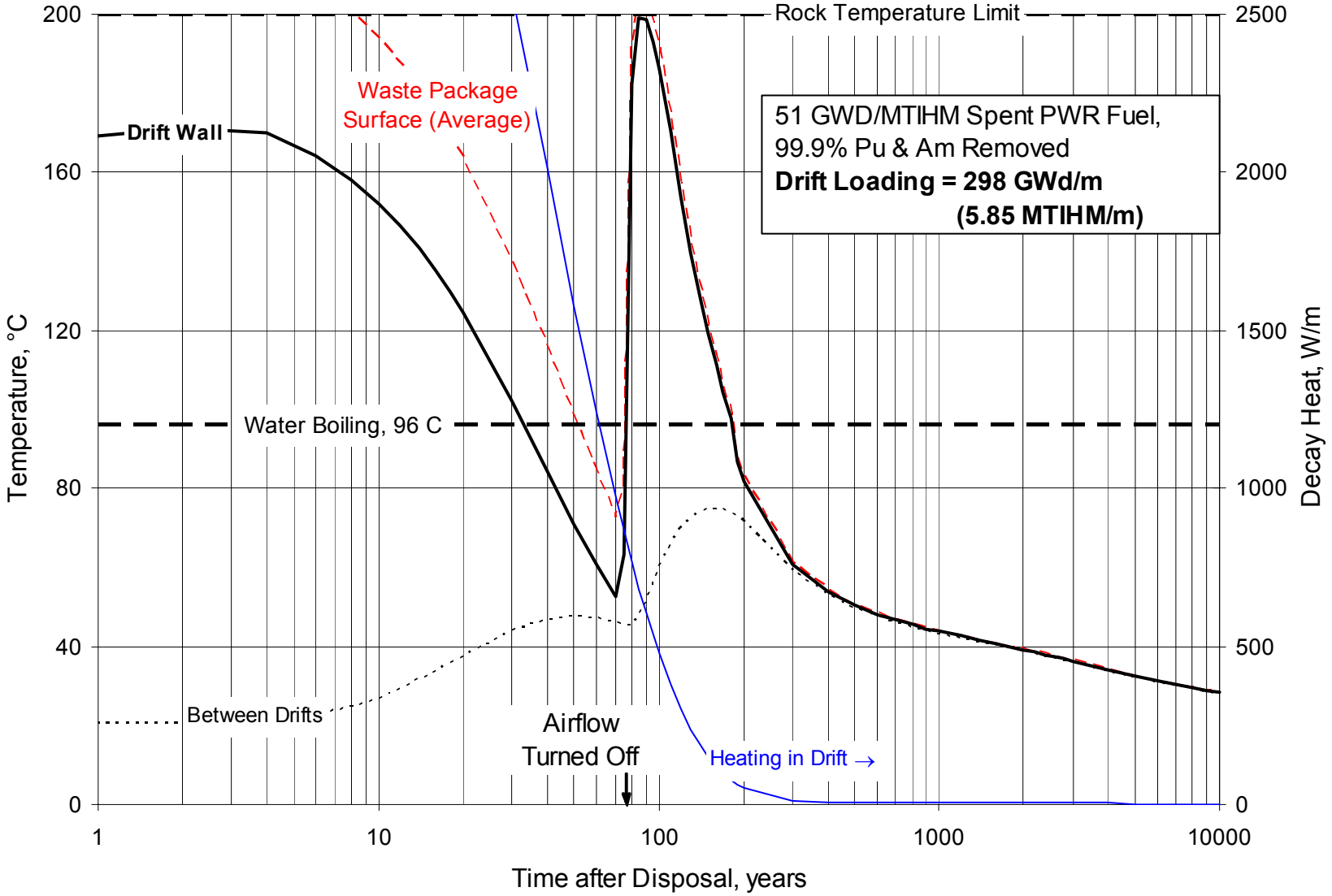
Repository Transient Thermal Response



Contributors to Spent PWR Fuel Decay Heat

- **The dominant contributors to the decay heat from 100 years to 1600 years are actinide elements**
  - most of the integrated decay heat is from Am241
    - *created by the decay of Pu241 in storage (14.4 year half-life)*
  - most of the rest of the decay heat is from isotopes of plutonium
  - fission products are relatively unimportant, providing only 32% of the decay heat at the time of repository closure
    - *fission products provide less than 6% by 200 years*
- **To have an impact on changing the drift loading in the repository, it is therefore necessary to remove the americium and plutonium from the spent PWR fuel**
  - if 99.9% of the Pu and Am were removed, the allowable drift loading could be increased, storing the waste from processing 5.85 MTIHM per meter of drift, representing material that generated 298 GWd of energy being stored per meter of drift

Spent PWR Fuel with 99.9% Pu & Am Removed



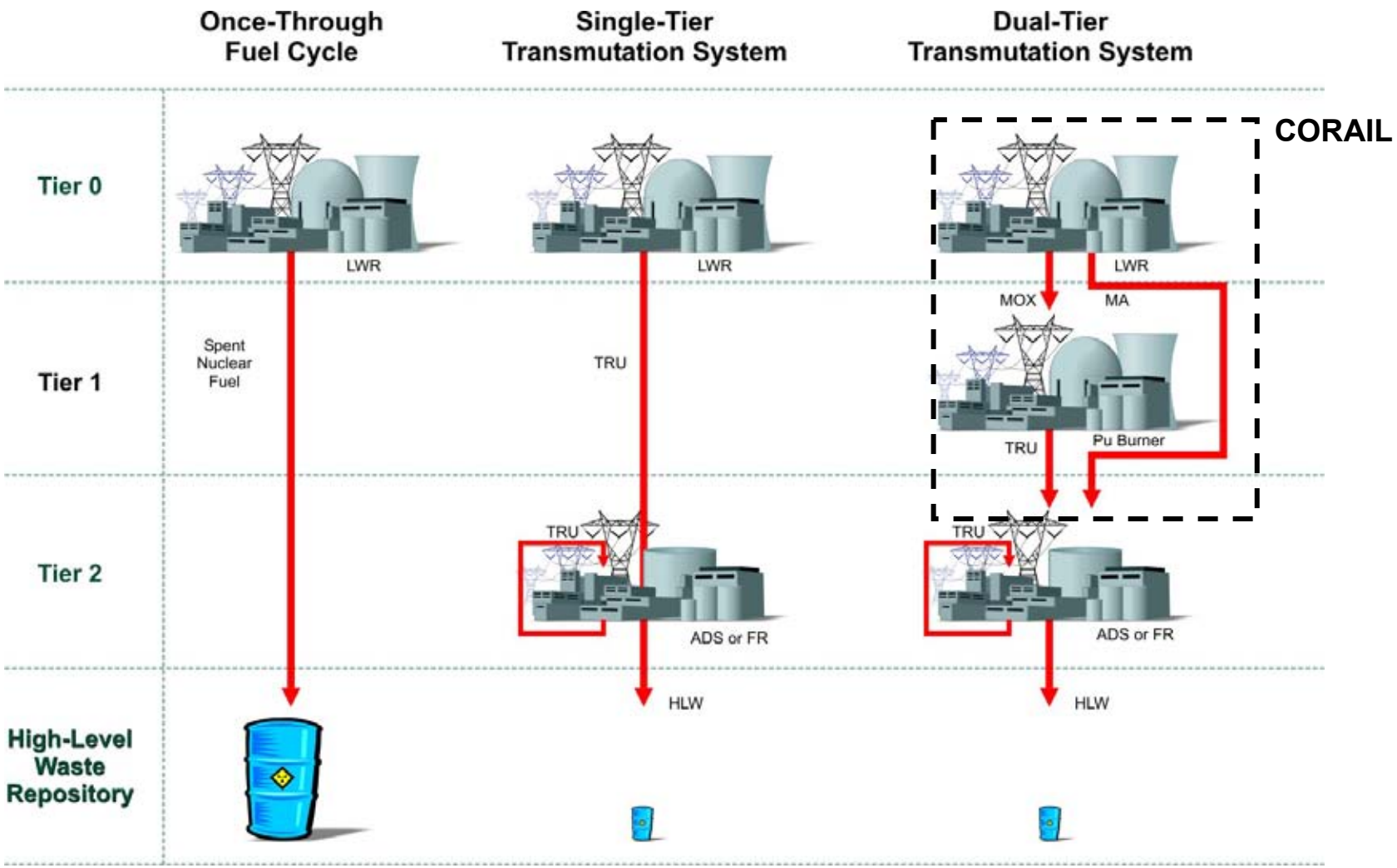
Repository Loading Benefits

- **With the Pu & Am removed, the drift loading can be increased by a factor of 5.1 relative to direct disposal of spent PWR fuel**
  - the peak temperature midway between adjacent drifts is no longer controlling the drift loading
  - the temperature limit of 200 ºC for the drift wall now determines the drift loading
- **Since the drift wall temperature peaks between 75 and 85 years**
  - Dominant isotopes contributing to decay heat in this time period are cesium and strontium fission products
- **With the Cs & Sr also removed, the drift loading can be increased by a factor of 51 relative to direct disposal**
  - the temperature limit of 200 ºC for the drift wall still determines the drift loading, but occurs at the time of placement
  - Removal of Cm could result in a factor of about 100

Transmutation Approach for Improved Waste Management

- **Long-term heat, radiotoxicity, and dose are all dominated by the Am-241 to Np-237 decay chain**
- **Destruction of the transuranics (TRU) is targeted to eliminate the problematic isotopes**
- **Some form of reprocessing is necessary to extract transuranic elements for consumption elsewhere**
- **The transuranic (TRU) inventory is reduced by fission**
  - Commonly referred to as 'actinide burning'
  - Transmutation by neutron irradiation
  - Additional fission products are produced
- **In the interim, the TRU inventory is contained in the transmutation fuel cycle**

Transmutation System Approach

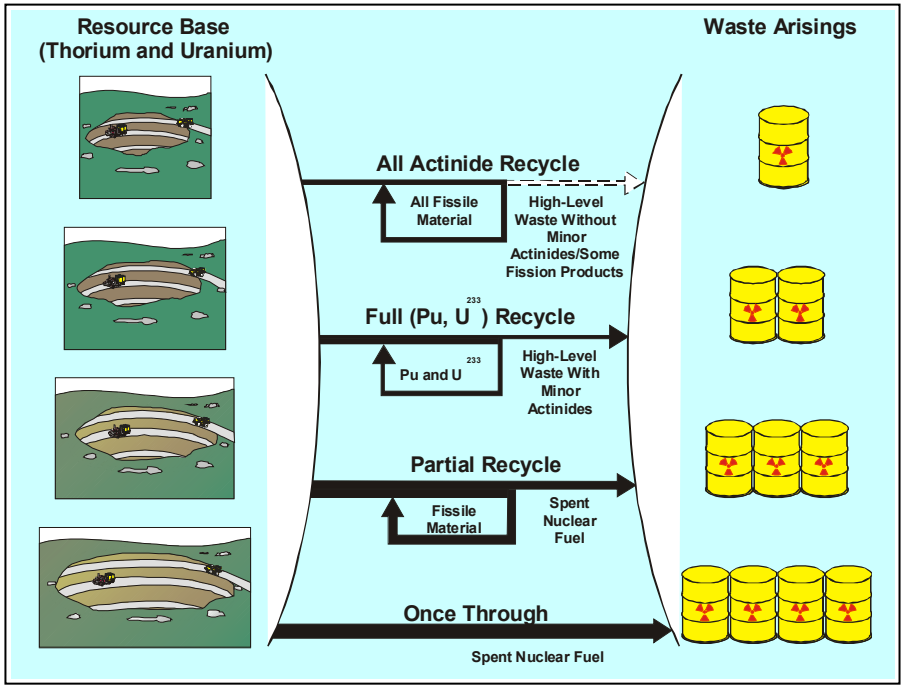


Generation IV Fuel Cycle Studies

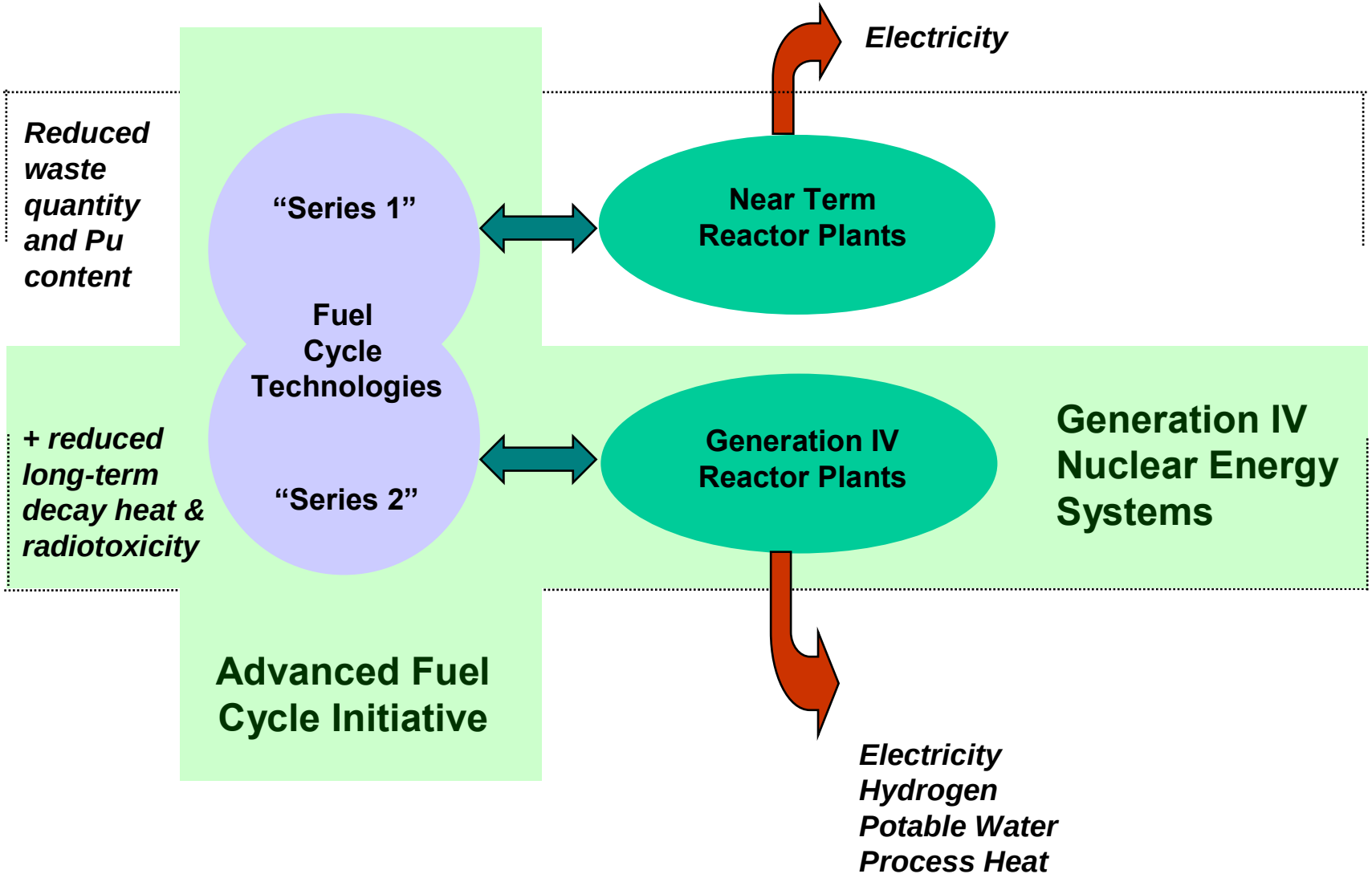
- **Focused on waste generation, <u>resource consumption</u>, and fuel cycle cost over the Generation IV planning horizon (→ 2100)**
- **Based on the projected growth in world demand for nuclear energy, nominally WEC/IIASA "middle course" scenario**

  Now: 350 GWe
  2050: 2000 GWe
  2100: ~6000 GWe

- **Evaluations represented**
  - Transitions from current fleet
  - Symbiotic deployment of systems fulfilling different missions
- **Scenarios run for generic fuel cycles**
  - Once-through (baseline case)
  - Partial fissile recycle
  - Full fissile recycle
  - Full actinide recycle

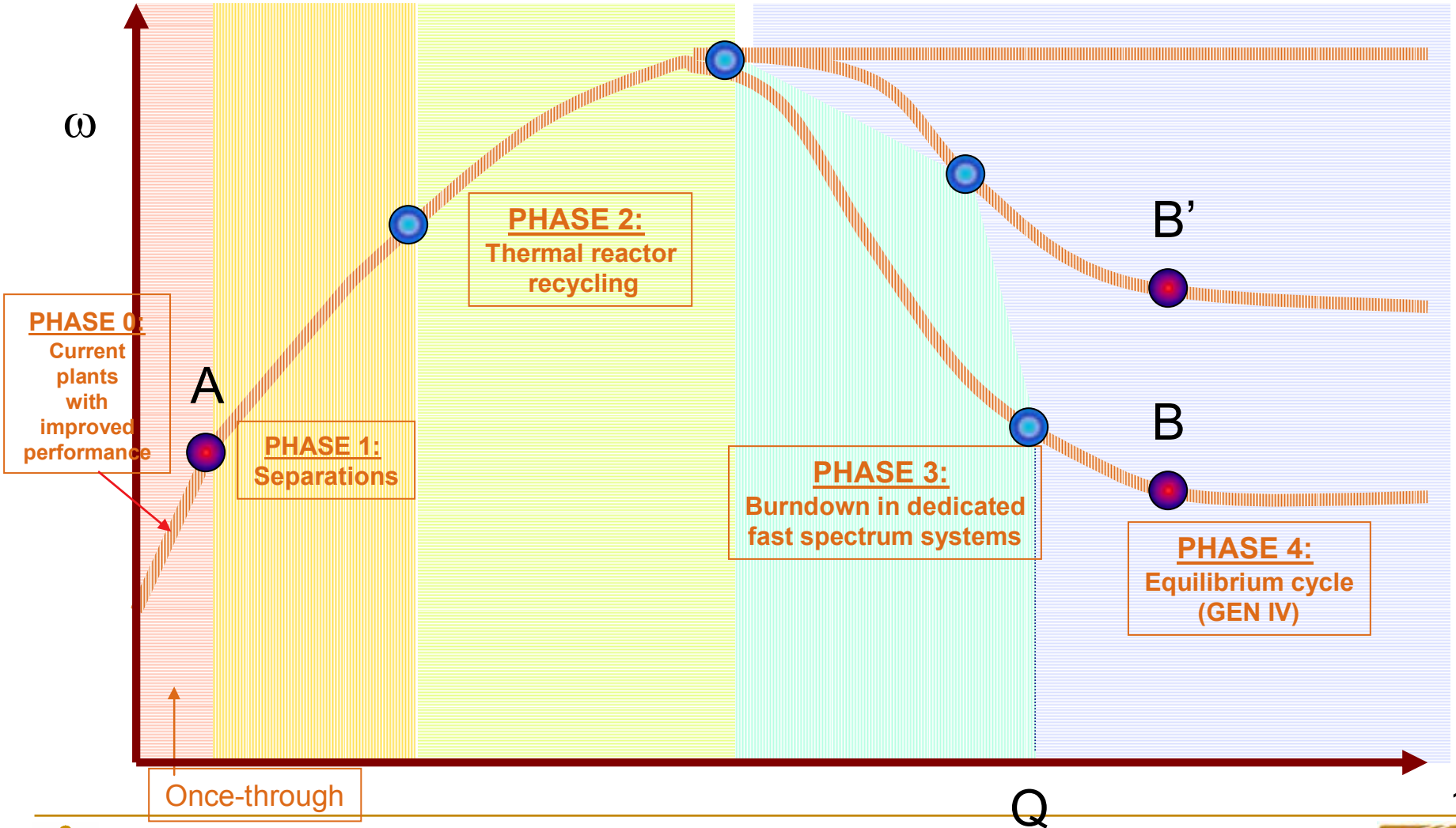


Relation of Advanced Nuclear Reactor and Fuel Cycle Programs



Potential Phases for AFC Implementation (from AFCI Program Plan)

ω = F(volume, radiological risk, short-term heat load, long-term heat load,plutonium mine)



Discussion of AFCI Strategic Framework

- **Phases represent major changes in fuel cycle approach**
  - Implementation of processing, recycle, advanced reactors
- **Timing and duration of phases will vary greatly depending on both *policy* and *technology* decisions**
  - Existing infrastructure will slow the transition between phases
    - *For example, reactors not replaced until end of design life*
  - Phases may be skipped completely if next step is ready
  - Waste management within each phase will be dictated by consideration of subsequent phases - *not limit future options*
- **Performance will vary between fuel cycle metrics (e.g., repository capacity vs. plutonium inventory)**
- **Flexibility to accommodate alternate futures is critical**
  - Large uncertainty in nuclear power growth or decline
  - Infrastructure development requires long lead time
  - Timing and technology of Generation-IV not yet specified

Advanced Fuel Cycle Transition Scenarios

What is the transition from the current once-through LWR fuel cycle to a sustainable Generation-IV nuclear fuel cycle?

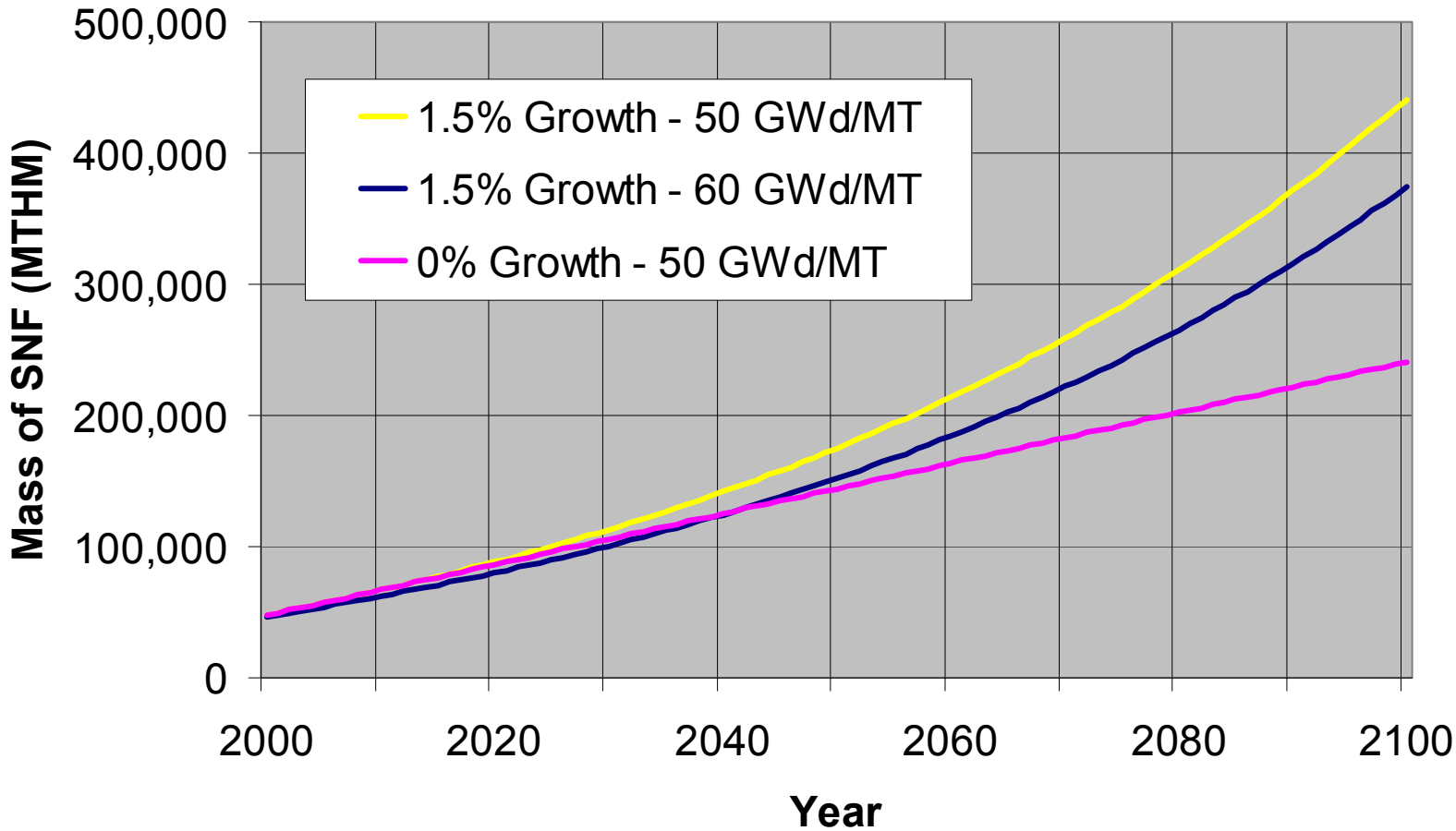
- Depends intimately on nuclear future assumptions
- Infrastructure requirements – reactors and processing
  - *Introduction and replacement rates may dictate timing*
- Variety of technology options must be encompassed
  - *Economics and external factors may dictate choices*

- **Illustrative fuel cycle scenarios**
  - Constant power with implementation of advanced fuel cycle
  - Nuclear growth with aggressive gas reactor introduction
- **Comparison of equilibrium end states**
  - Evolution of technology options
  - Relation to Generation-IV candidates

Illustrative Scenario #1: "Summer Study" 2003

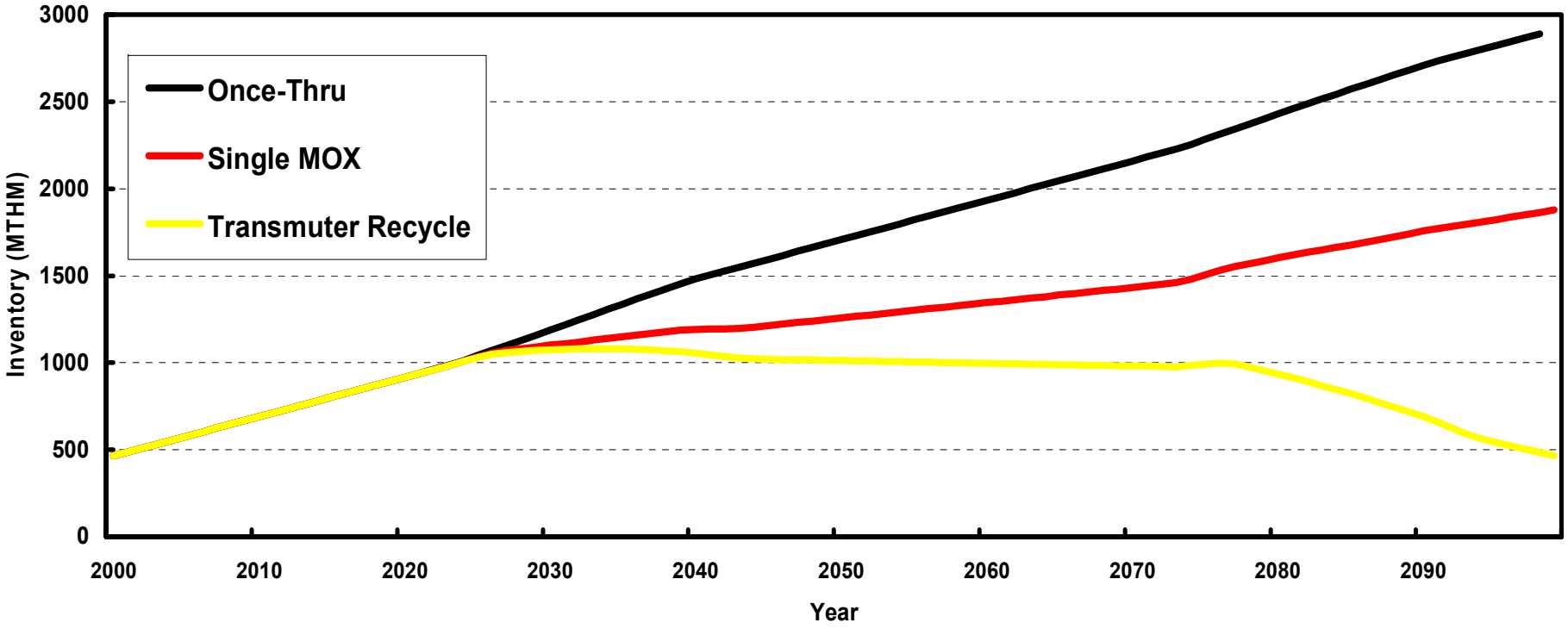
- **Results summarized at September 2003 ANTT meeting**
- **Fuel cycle impact evaluated for limited set of scenarios**
  - Once-through and separations only
  - Single MOX recycle
  - Single and double tier transmutations systems
- **Dynamic analysis of fuel cycle performance**
  - Considered stable and growth scenarios
  - Estimate of infrastructure requirements
  - Impact of reprocessing on spent fuel characteristics
  - Tracking of material inventories throughout entire fuel cycle

Evolution of Spent Fuel Inventory in Scenario #1



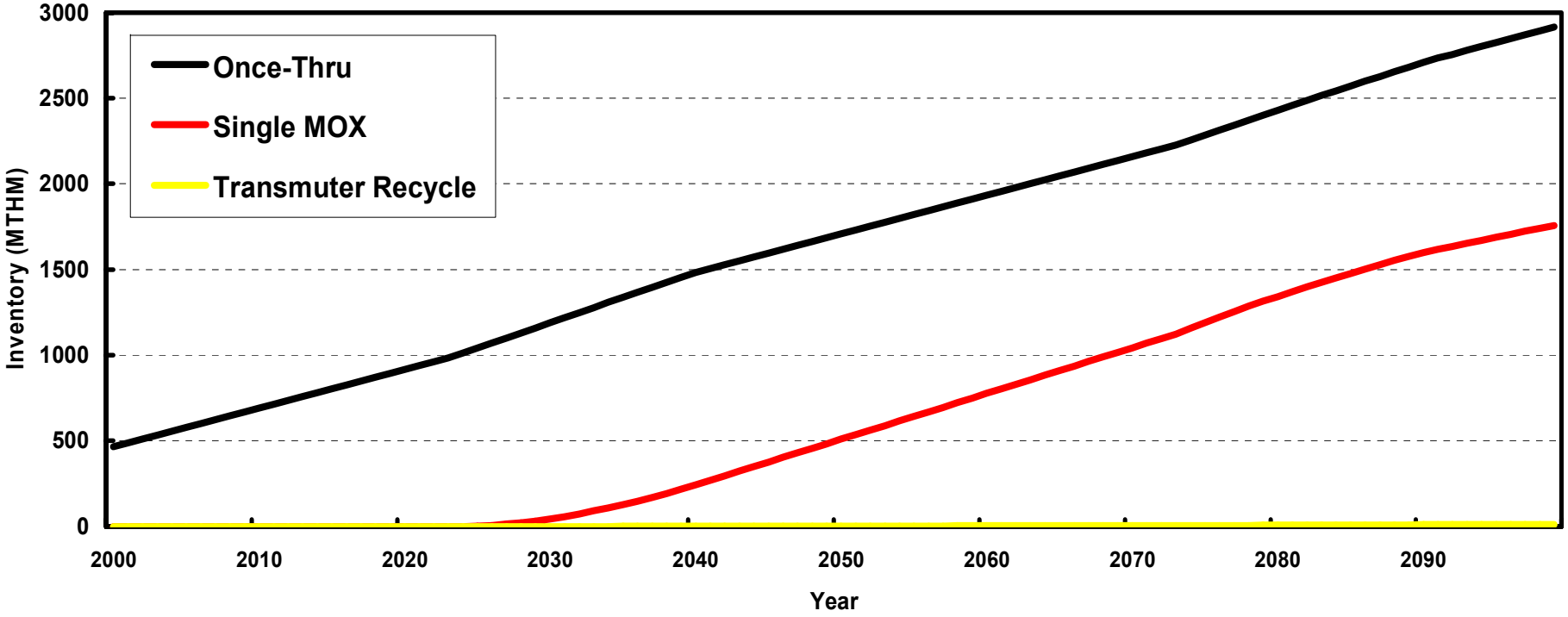
- **Even modest growth can double inventory by 2100**
- **Increased LWR burnup can slow SNF and Pu inventory growth rate**
  - However, minor actinide inventory will not be reduced

Evolution of Plutonium Inventory in Scenario #1



- **MOX recycle can stabilize the plutonium inventory**
  - However, minor actinide inventory steadily increases
- **Plutonium inventory can be reduced in multi-recycle scenarios**
  - Inventory management requires significant burner capacity
  - Not significant capacity until 2080

Plutonium Inventory Disposed to Waste in Scenario #1



- **Important to distinguish location of plutonium inventory**
  - Destined for waste in once-through fuel cycle
  - Single cycle delay and then to waste for MOX
  - Contained in processing/reactor for multi-recycle

Illustrative Scenario #2: Aggressive Deployment of Gas Reactors

Nuclear Generating Capacity Deployment Schedule

Present

100 GWe (Total Capacity = 909 GWe, or 11% nuclear)

Present - 2024

Total Capacity grows by 21% to 1100 GWe. Nuclear capacity increases to 103.4 GWe (9.4% of Total), but growth in nuclear doesn't begin until 2015.

2025 - 2029

103.4 GWe, no growth in nuclear or Total capacity

2030 - 2050

Total Capacity increases by 1%/year. Nuclear share of Total increases 0.5%/year, reaching 19.9% in 2050.

Fuels/Systems Deployment Schedule

Present - 2014

High-burnup (50 GWd/MT) UO_2 fuel in an all LWR fleet

2010 - 2014

Begin retirement of LWRs at rate of 0.25 GWe/year.

2015 - 2029

Continue retirement of LWRs at rate of 0.25 GWe/year and replace with ALWRs fueled with ultra-high burnup UO_2 (100 GWd/MT).

2015 - 2019

Transition all remaining LWRs to ultra-high burnup UO_2.

2025 - 2029

Transition remaining LWRs and all ALWRs to partial-core loading of Pu-based NFF (30%) and ultra-high burnup UO_2 (70%), *but limited by a 3,000 MT/y UO_2 processing capacity.*

2030 - 2034

Continue retirement of LWRs at an accelerated rate of 4.75 GWe/year replace with UO_2-fueled GT-MHRs.

2035 - 2049

Complete retirement of LWRs at 4.75 GWe/year, replacing retiring LWRs with *either UO_2-fueled GT-MHRs or, if spent UO_2 processing capacity is available, GT-MHR Pu-burners*.

2030 - 2039

Increased nuclear generating capacity by addition of GT-MHRs.

2040 - 2050

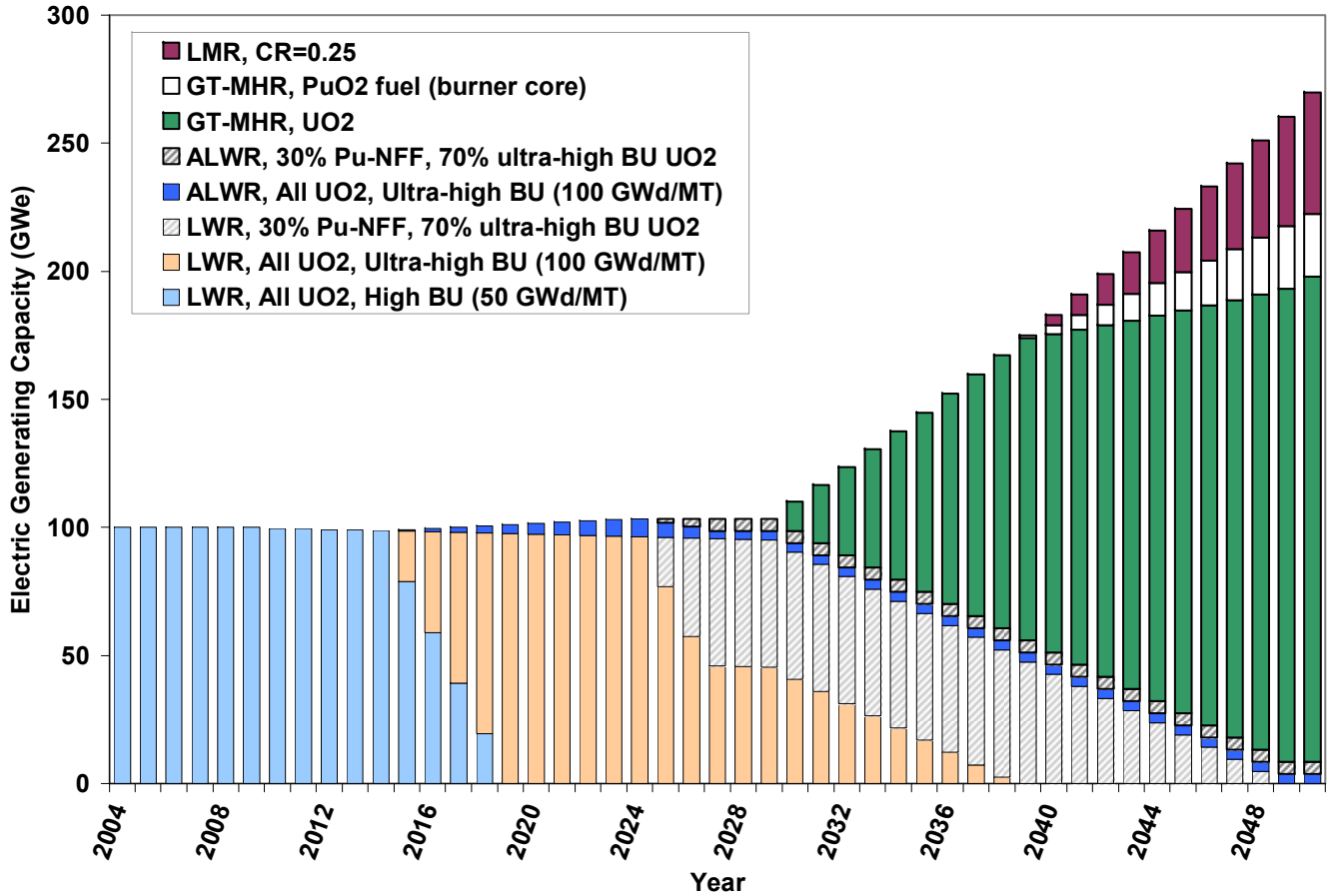
Increased generating capacity accommodated by equal addition of GT-MHRs and low conversion ratio fast reactors (LMRs).

Fuel Cycle Parameters for Scenario #2

- **Key data compiled from previous AFCI fuel cycle analyses**
  - LWR, gas reactor and fast reactor (SFR) options considered
  - Explicitly track spent fuel mass, plutonium and MA inventories

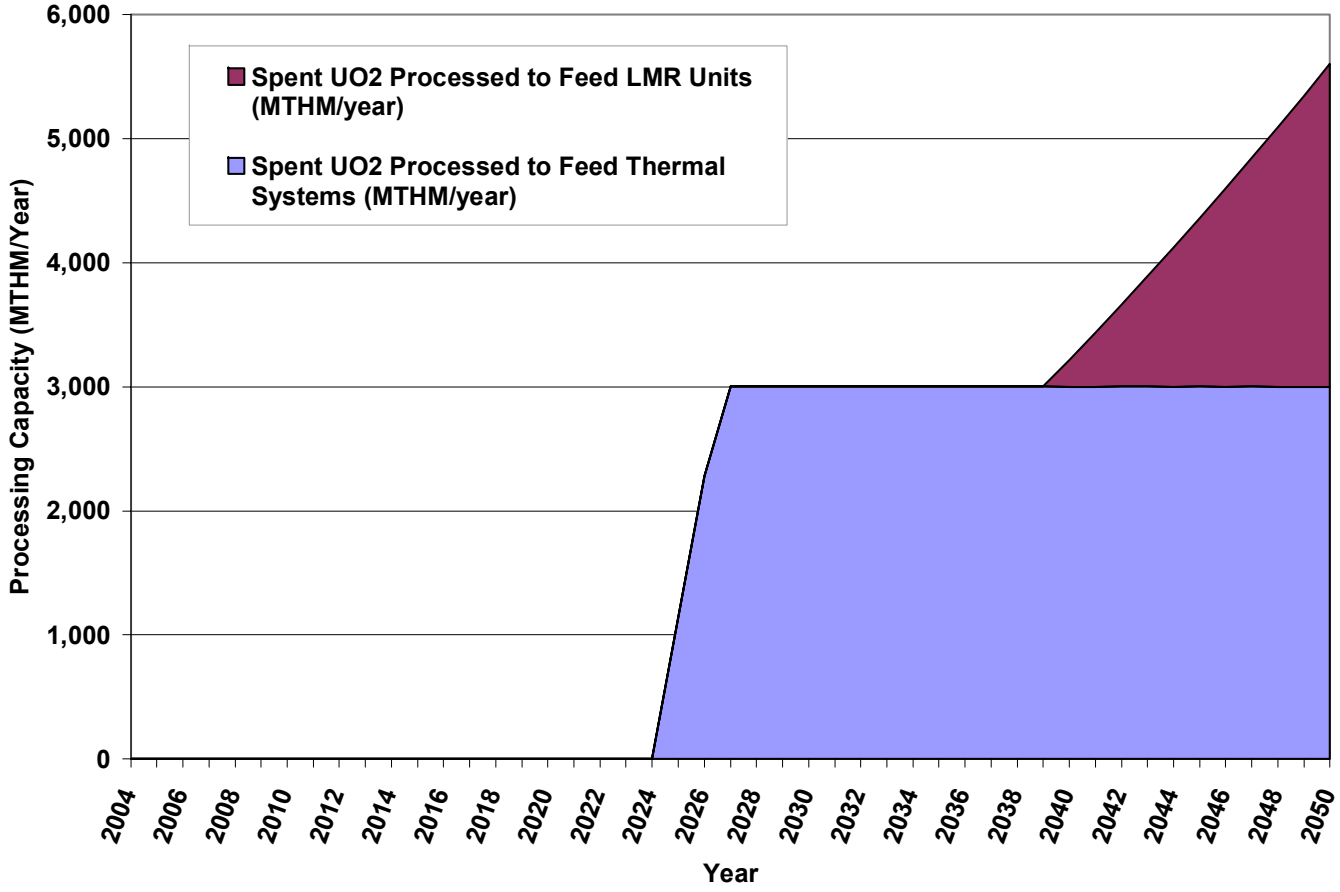
Reactor System	Unit power (GWth)	Thermal to electric conversion efficiency	Fuel form	Fuel discharge burnup (GWd/MT)	Net production at fuel discharge (g/GWhre)						Reactor-Grade Pu or TRU Feed Rates	
					Pu	***MA***	***Pu-238***	***Pu-241***	***Am-241***	***Cm-244***	g/GWhre	MT/GWe/yr
LWR (Thermal)	3	33%	UO_2	33	34.89	2.79	0.53	4.74	0.14	0.10		
				50	32.07	3.79	1.00	4.02	0.15	0.26		
				100	20.96	3.28	1.27	3.55	0.17	0.44		
			MOX (Pu+Np)	44.4	-46.68	3.21	6.16	8.77	2.71	1.93		
			IMF (PuO_2)	510	-138.8	11.14	-2.38	3.55	1.70	3.37	246.80	1.84
			IMF ($TRU\text{-}O_2$)	510	-117.0	-10.35	8.74	1.02	-9.21	4.70	245.85	1.83
GT-MHR[2] (Thermal)	0.6	47%	UO_2[3]	*100*	*14.72*	*2.30*	*0.89*	*2.49*	*0.12*	*0.31*		
			$PuO_{1.7}$	591	-91.05	4.95	-1.81	-1.62	1.35	3.48	150.00	1.12
			$TRU\text{-}O_{1.7}$	470	-81.92	-4.85	6.12	0.32	-5.62	5.21	188.63	1.41
LMR (Fast), CR=0.25, OT cycle	0.84	38%	U/TRU/Zr	177	-70.85	-11.06	3.73	-8.31	-4.53	0.65		0.60[4]
LMR (Fast), CR=1.0, Eq. recycle[5]		38%		*150*	*0*	*0*	*0*	*0*	*0*	*0*		

Evolution of Nuclear Power Capacity in Scenario #2



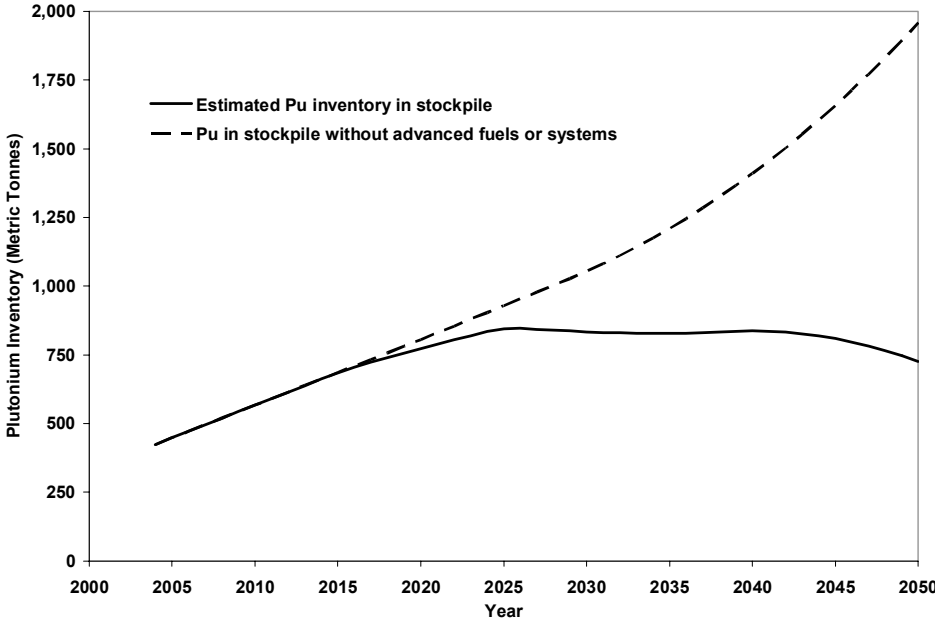
- **LWRs transition to high burnup and then IMF fuels**
  - Begin to retire in 2015, replaced by ALWRs, then GT-MHRs
- **Large capacity of GT-MHRs (~200 GWe) to meet growth**

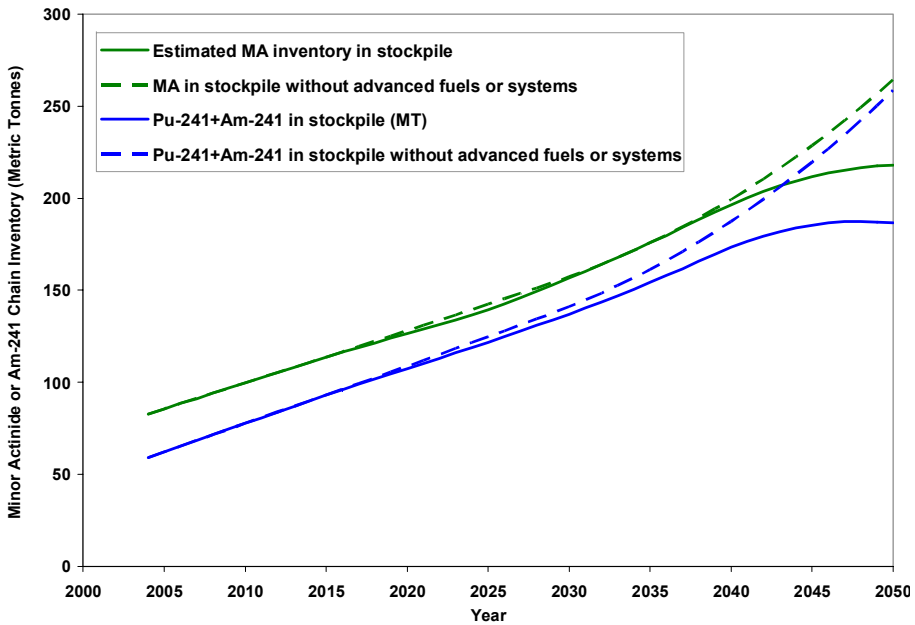
Evolution of Processing Capacity in Scenario #2



- **The number of thermal Pu-burners is constrained by 3,000 MT/yr rate**
- **LMR units are low conversion ratio burners**
  - Their make-up feed requires significant increase of processing capacity

Evolution of Plutonium and MA Inventories in Scenario #2





- **Without AFC, Pu and MA inventory would increase by 4X by 2050**
- **Thermal Pu burners limit the increase of the plutonium inventory**
  - Capped at level of ~750 MT when processing begins
  - Pu inventory begins to decrease when LMRs introduced in 2040
- **However, MA inventory is not stabilized until FRs introduced**
  - Reflects thermal recycle assumptions in this particular scenario

Illustrative Scenario #2: Trends and Conclusions

- **For growth scenario, a massive infrastructure is required**
  - Nuclear capacity increased to 250 GWe by 2050
    - *Current LWR fleet must be replaced in same time frame*
  - If available, large capacity of advanced reactors employed
    - *In this scenario, ~200 GWe of GT-MHRs*
  - Processing capacity of 3,000 MT increasing to 5,000 MT/yr
    - *Backlog of LWR SNF not consumed by 2050*
    - *Limits of rate of Pu-burner fuel introduction*
    - *Large increase required for LMR burner support*
- **Impact on waste inventories is modest**
  - Plutonium inventory stabilized by thermal burners
  - Minor actinide inventory requires recycle system (LMRs)
- **Thus, to achieve the desired waste benefits many burner systems are needed, and long lead time will be required**

Continuous Recycle Options

- **A variety of technology options can be employed**
  - Rough equilibrium estimates from previous AFCI systems studies

Transmutation System	Power Fraction of Thermal UO2	Power Fraction of Thermal MOX/IMF	Power Fraction of Fast Reactors
Continuous Recycle MOX	68	32	0
Continuous Recycle IMF	88	12	0
Dual Strata	65	15	20
Single Strata	70	0	30
Sustained Fast Reactor	0	0	100

- **Ratio can be varied to manage plutonium inventory**
- **Different strategies may be employed as new technology for fuels and reactors is introduced**

Transmutation Performance Expectations for Generation-IV Systems

- **Two thermal reactor options (VHTR gas reactor and supercritical water reactor, SCWR)**
  - To first order will behave like ALWR systems
    - *Will require enriched uranium for continuous recycle*
    - *Will need curium segregation and storage*
  - Detailed transmutation performance of VHTR, as replacement for LWRs as UO2 system is being evaluated
  - SCWR has fast spectrum option, with additional R&D needs
- **Three fast reactor options (sodium-cooled, SFR; lead-cooled, SFR, and gas-cooled, GFR)**
  - To first order, will behave like SFR (detailed closed cycle studies)
  - Initial study of GFR burner design yielded similar results
- **Molten salt reactor (MSR) is a unique case**
  - Default fuel management is continuous recycle
  - More R&D required than other options

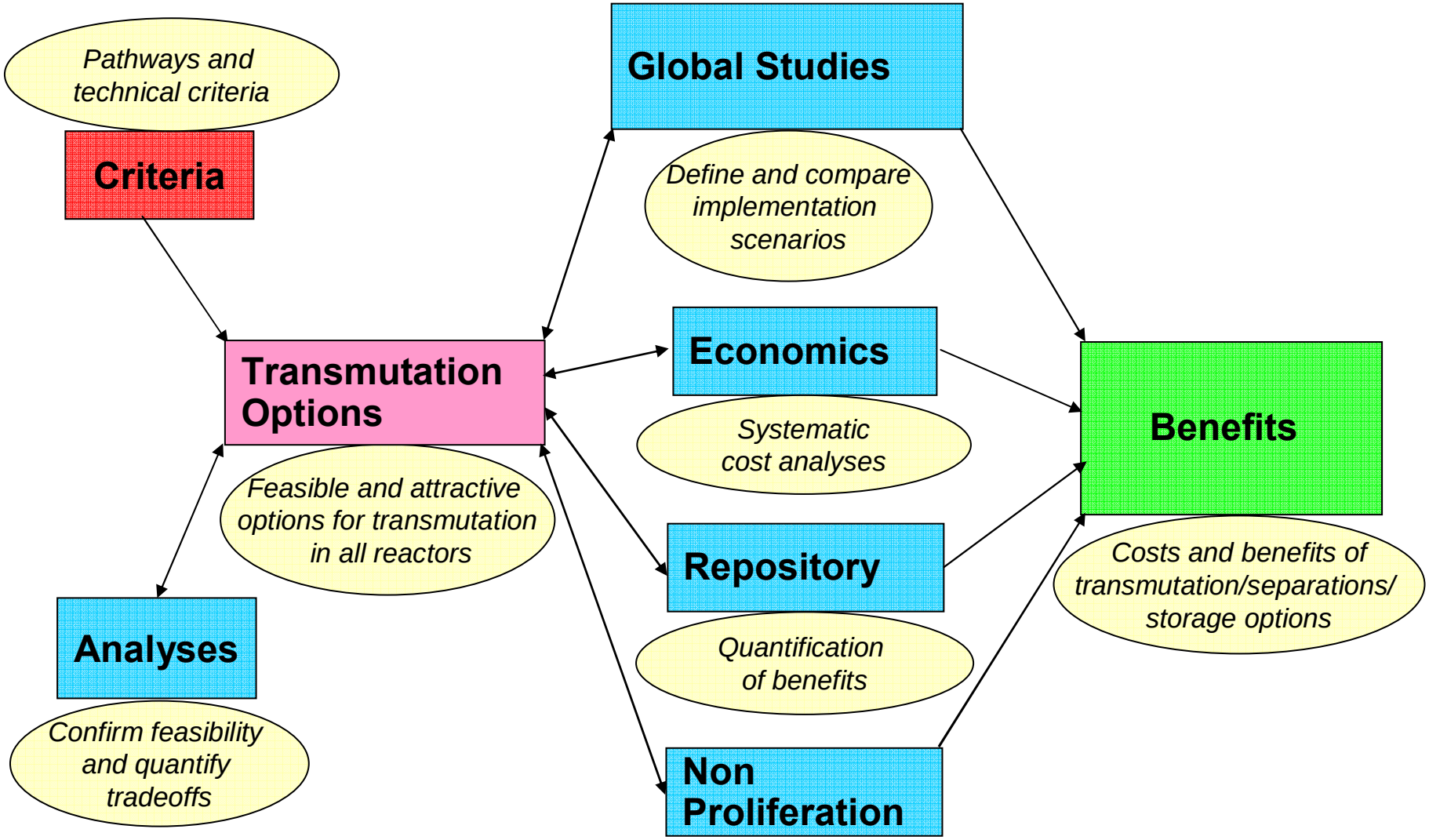
Conclusions on Advanced Fuel Cycle Transition

- **To achieve large repository benefits, eventual goal must be introduction of a continuous recycle technology**
  - Implies large-scale processing and utilization of advanced fuels
- **In short-term (decades), LWRs will be the dominant reactor type**
  - Separations technology can be developed and implemented
  - Transmutation fuels (either MOX or IMF) can be developed
  - Recycle can be initiated with some benefit
- **Continuous recycle in thermal reactors has known limits**
  - Reliance on enriched uranium support
  - Buildup of higher actinides (Cm separation, handling issues)
  - Safety issues for high plutonium loadings (burner concepts)
- **Therefore, Generation-IV fast reactors remain an attractive solution**
  - Inhibit production of higher actinides, can burn all transuranics together
  - Future utilization of all uranium resources without enrichment
- **Combined fuel cycles exist allowing gradual transition from thermal to fast systems with sustained waste management benefits**

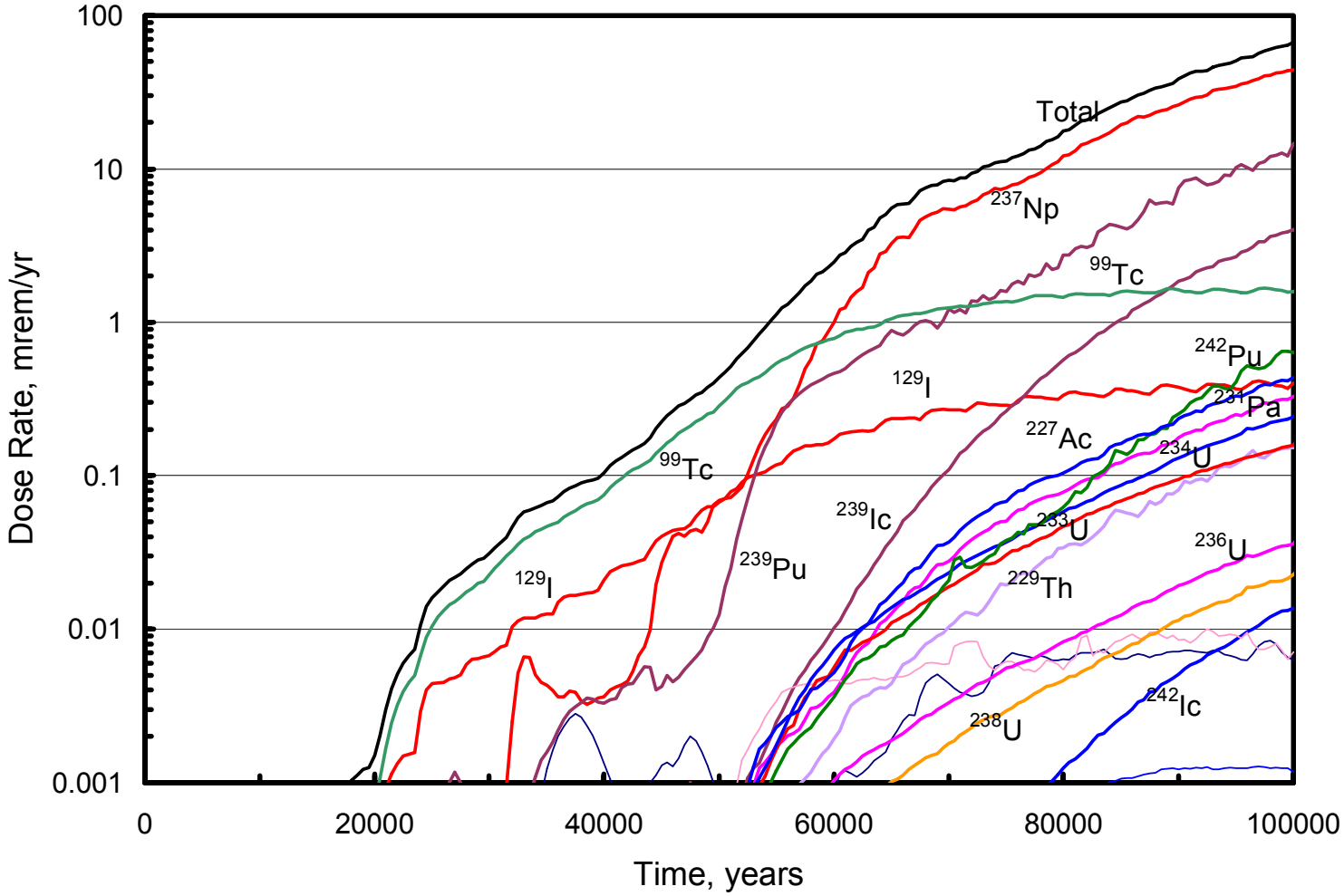
BACKUP

- AFCI Systems Analysis Strategy - 1
- Short Term YMP Released Dose Rates - 1
- Multi-tier Study Radiotoxicity Impact - 1
- YM Thermal Model and Rep Benefit Details – 7
- Advanced Safeguards - 1
- LWR Recycle Nonproliferation Comparison – 1
- Transmutation Potential – 4
- LWR Transmutation Options – 16
- FR Transmutation Options - 7

AFCI Systems Analysis Strategy

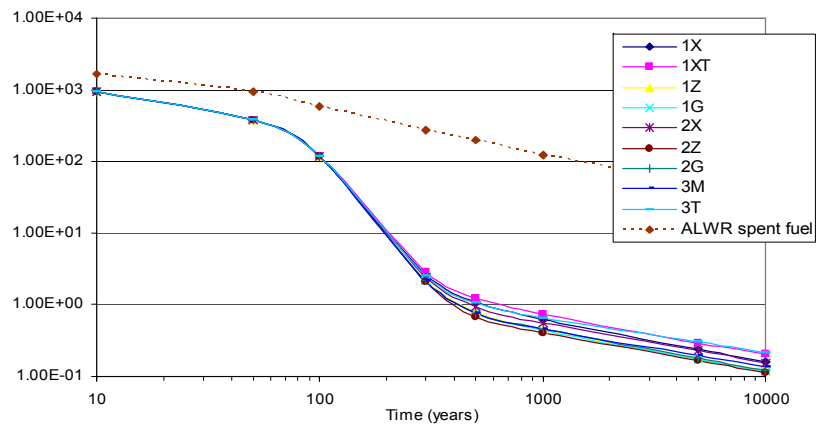


Estimated Dose Rates for Direct Disposal – Short-term

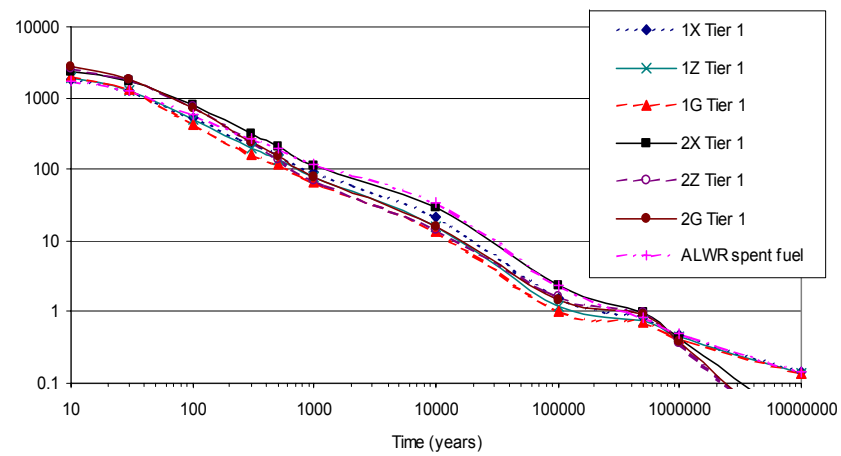


Key Conclusions from FY2001 Multi-Tier Fuel Cycle Study

- **Given clean fuel processing (0.1% losses), typical goals for transmutation can be achieved**
  - TRU and plutonium losses to waste less than 0.6%
  - Radiotoxicity below level of natural ore in < 1,000 years
- **First tier thermal spectrum irradiation does not significantly reduce the radiotoxicity**
  - Confirms need for a final tier fast spectrum system
- **Utilization of first tier thermal spectrum system can increase the Tier 2 support ratio**
  - Fewer specialized transmutation systems required

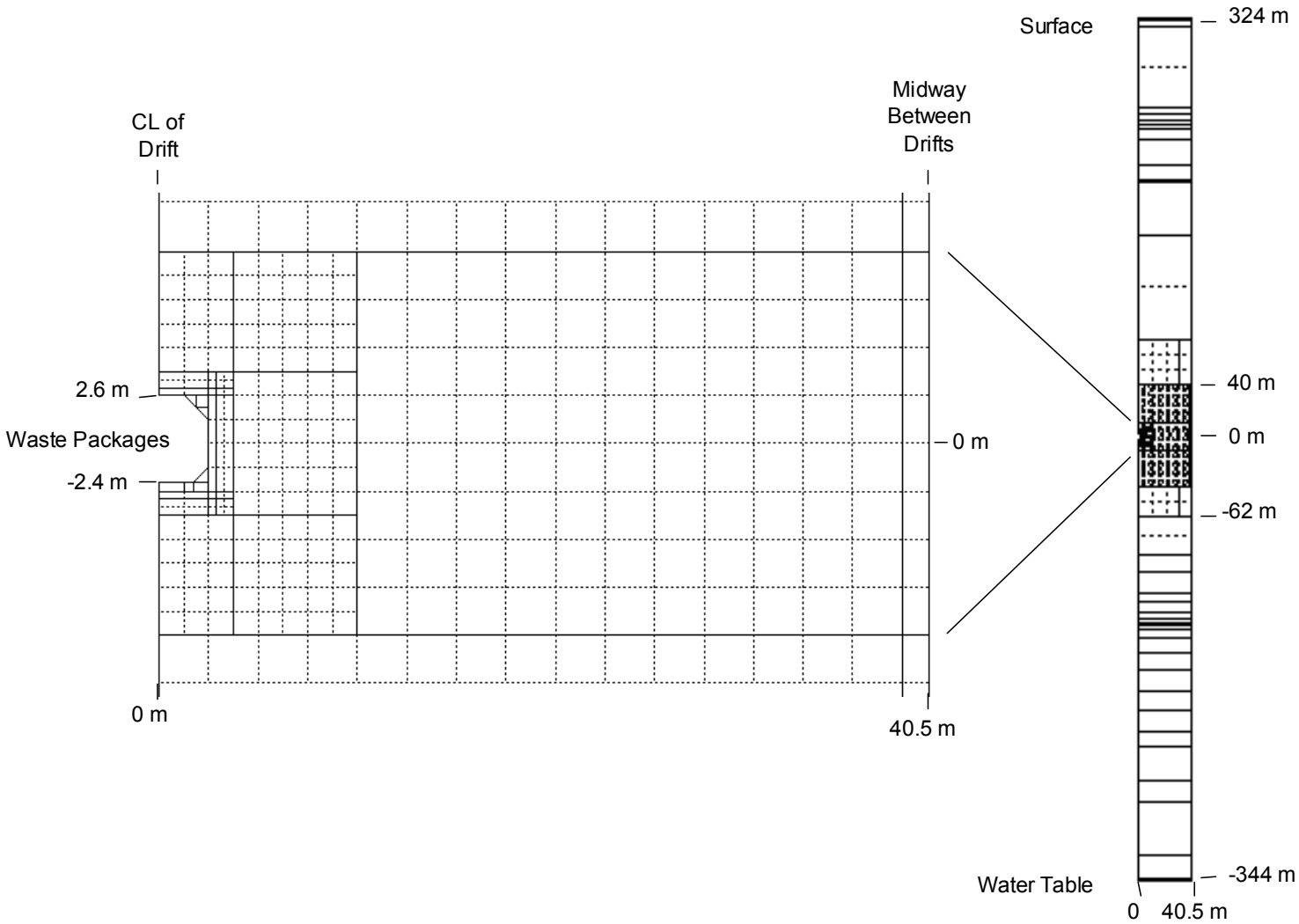


Double Tier Irradiation

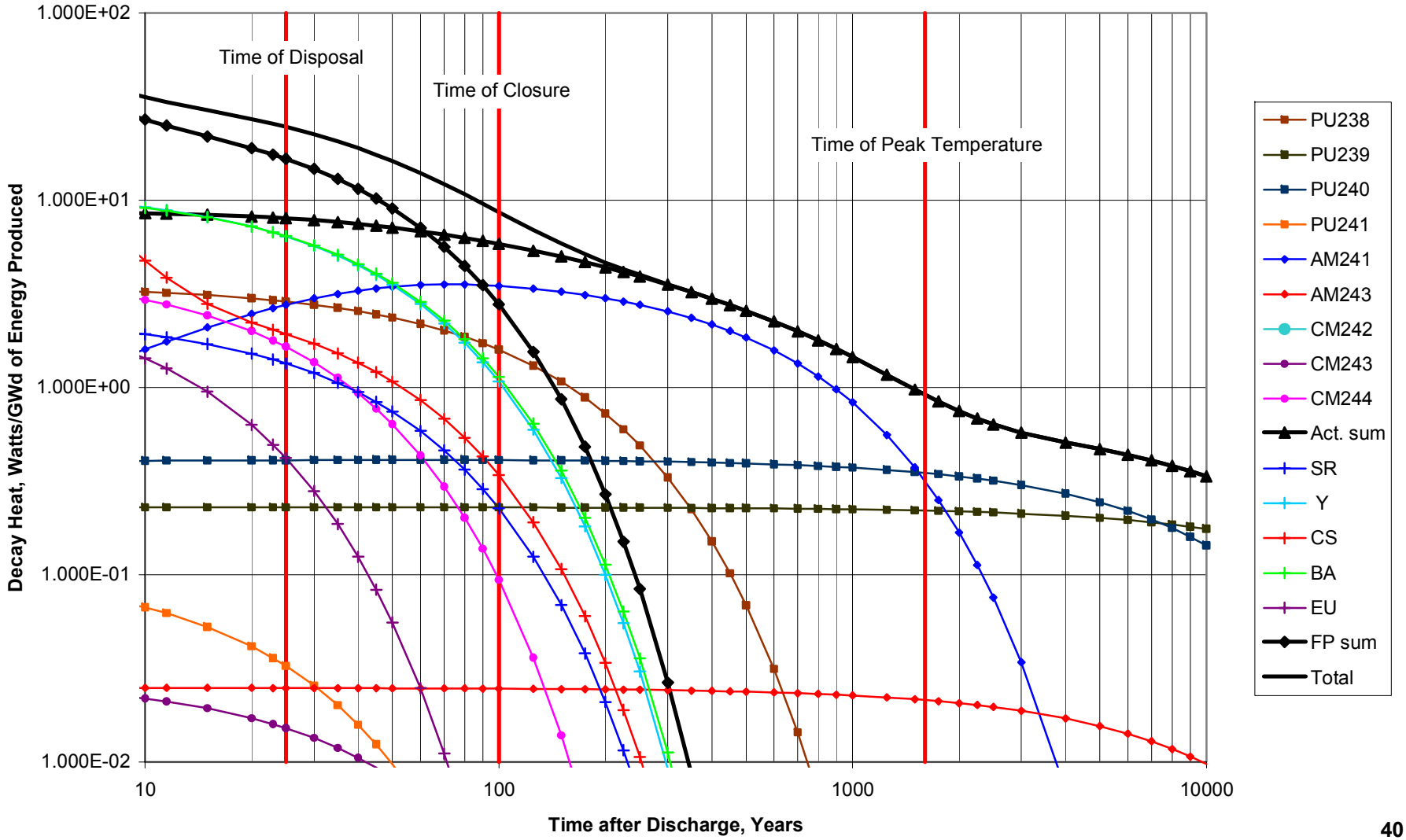


First Tier Irradiation Only

Yucca Mountain Thermal Model



Spent PWR Fuel Decay Heat

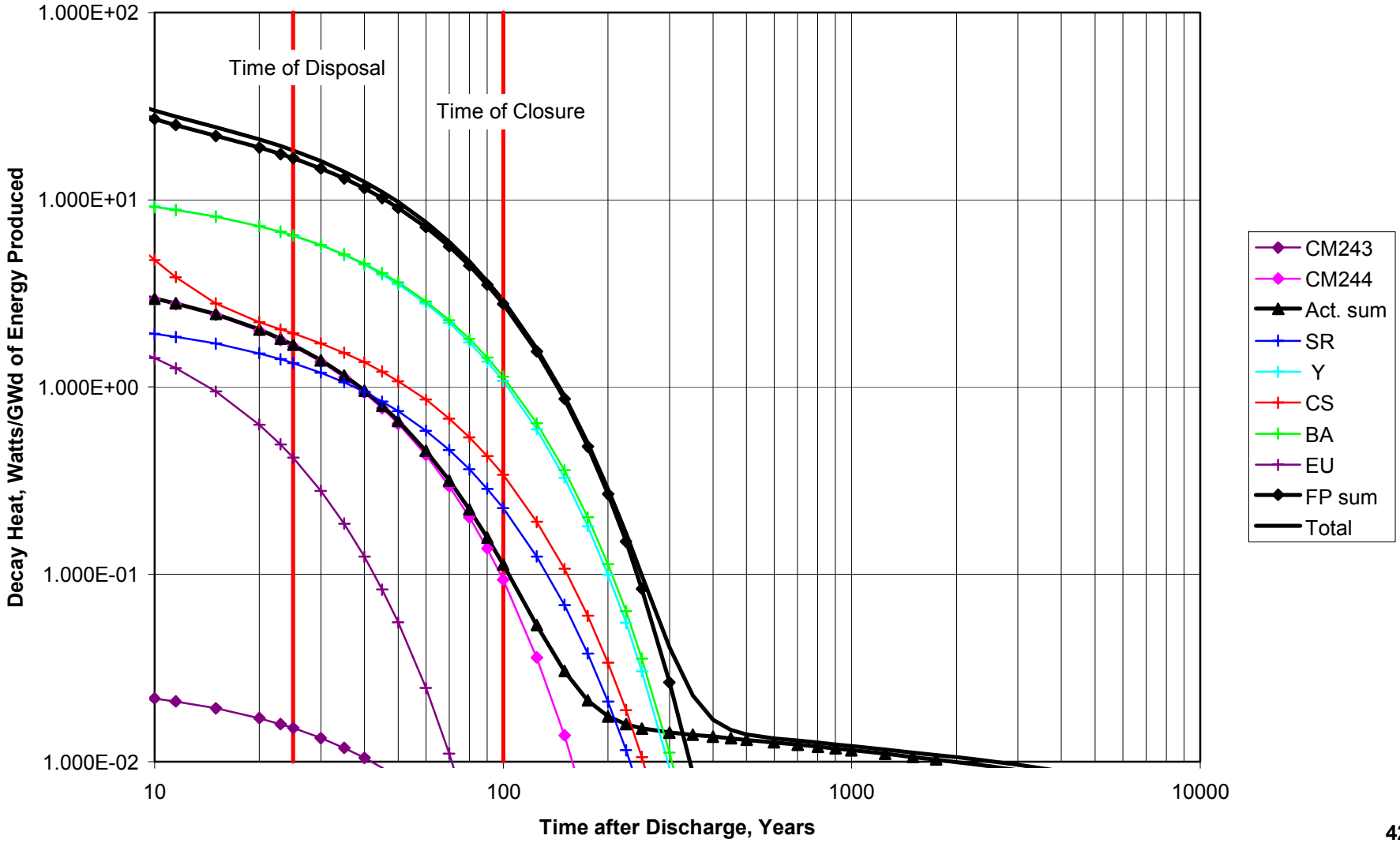


Drift Loading Limitation w/o Pu & Am

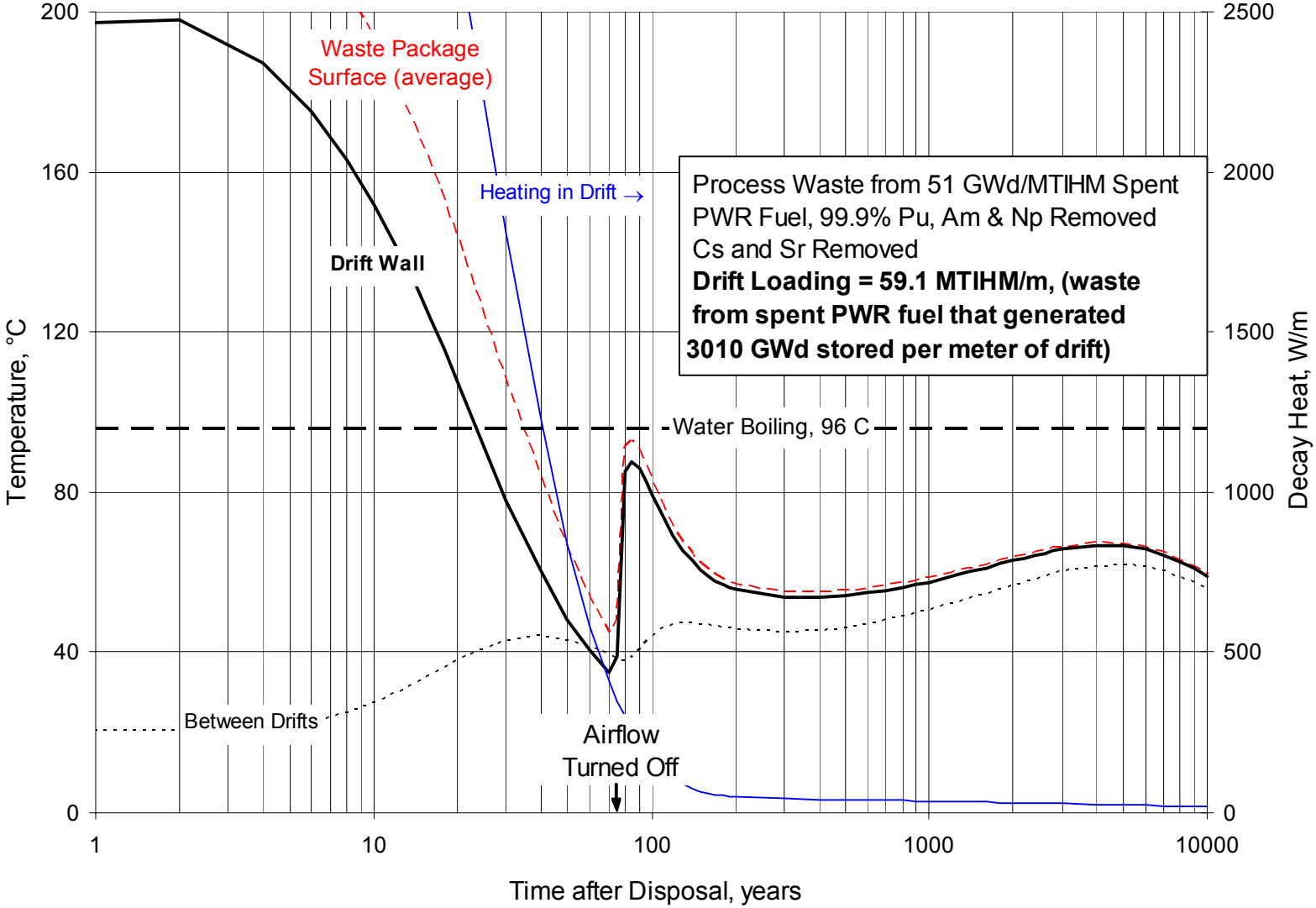
- **With the Pu & Am removed, the drift loading can be increased by a factor of 5.1 relative to direct disposal of spent PWR fuel**
  - the peak temperature midway between adjacent drifts is no longer controlling the drift loading
  - the temperature limit of 200 ºC for the drift wall now determines the drift loading
  - the temperature peak occurs at 85 years, shortly after the closure of the repository

- **Since the drift wall temperature was low prior to repository closure, the source of the decay heat is from isotopes with the highest decay heat between 75 and 85 years**
  - in the absence of plutonium and americium, the dominant isotopes contributing to decay heat in this time period are barium and yttrium, short-lived decay products of cesium and strontium

Spent PWR Fuel Decay Heat w/o Pu & Am



Spent PWR Fuel w/o Pu, Am, Cs & Sr



Drift Loading Limitation w/o Pu, Am, Cs & Sr

- **With the Pu, Am, Cs & Sr removed, the drift loading can be increased by a factor of 51 relative to direct disposal of spent PWR fuel**
  - the peak temperature of the drift wall after repository closure is no longer controlling the drift loading
  - the temperature limit of 200 ºC for the drift wall still determines the drift loading, but occurs at the time of placement
  - the source of the decay heat causing this temperature limit to be met is curium; removal of Cm results in a factor of about 100
- **The allowable increases in drift loading are representative of what is possible with reductions in Pu, Am, Cs & Sr**
  - Since the Pu and Am must be transmuted or fissioned, practical recycling schemes must be considered to determine the actual drift loading increases that can be realized
  - Cs and Sr will be sent to separate storage

Repository Performance Assessment

- **All aspects of repository performance must be considered**
  - Reactor physics analysis of once-through and recycle options
    - *All reactor types, including LWRs, gas-cooled reactors, and LMRs*
    - *Both fast and thermal spectrum systems*
    - *Detailed discharge compositions and rates for spent fuel and waste*
  - Modeling of reprocessing options
  - Thermal modeling of the Yucca Mountain Repository
    - *Detailed models of Yucca Mountain, including all heat transfer phenomena*
    - *Repository size is determined by heat load issues*
  - Total System Performance Assessment (TSPA)
    - *use the same GoldSim models as the Yucca Mountain Project (YMP) and the project reviewers*
    - *obtain estimates of peak released dose rate*

Development of Advanced Safeguards for Closed Fuel Cycle

- **Proliferation resistance of advanced fuel cycles is a key element of current programs (AFCI, GEN IV)**
  - The entire cycle must be included in the assessments (fuel fabrication, recycling, disposal,…)
- **Advanced design approach of pyroprocessing facility with <u>integrated safeguards</u> is being pursued**
- **System design for proliferation resistance includes intrinsic features as well as institutional measures, typical barriers being considered include:**
  - Materials properties
  - Difficult processes to modify, difficult to modify facilities
  - Increased automation
  - Process monitoring and intelligent software
  - Materials control and physical security
  - Increased transparency
  - International agreements
  - Multilateral inspections
  - Traditional safeguards

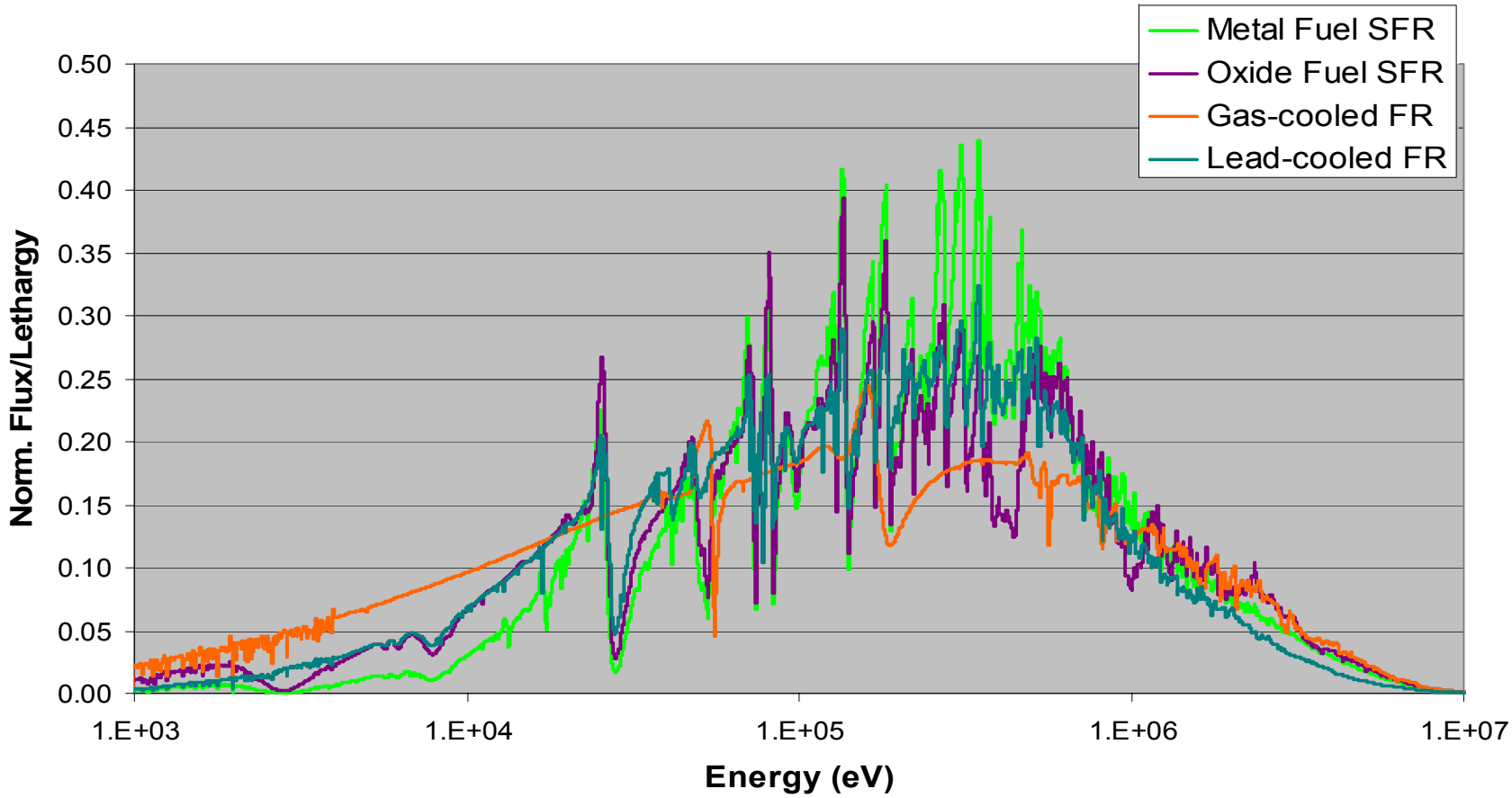
Summary of LWR Recycle Hypotheses Analysis Results

Recycle Hypothesis	Advantages	Disadvantages
Pu-only	Easiest to implement. Pu out of repository. Reduction in mid-term waste radiotoxicity and heat load.	Pu in fuel cycle needs safeguards (non-proliferation concerns). Radiotoxicity target unmet.
TRU	Clear benefits to repository. Provides time for advanced Series 2 systems to be deployed.	Fuel handling issues in fuel cycle. Limited recycles?
Pu+Np	Similar benefits to Pu-only case. With irradiation, Np-237 and a higher Pu-238 content provides marginal intrinsic radiation sources.	Does not significantly reduce Np-237 in repository. Radiotoxicity target unmet. Similar proliferation issues as Pu.
Pu+Np+Am (PNA)	Removal of Am-241 helps in the mid- and long-term (Np-237 minimized).	Presence of curium limits benefits to the repository (Pu-240 content) Fuel handling is a problem.
PNA and No Pu-242 or Pu242/Am-243	Radiotoxicity improved over PNA. Provides additional benefits to fuel handling over PNA.	Radiotoxicity target unmet.

Comparison of Transmutation Potential

- **Initial goal was to compare transmutation characteristics of the Gen-IV fast spectrum concepts – SFR, LFR, GFR**
  - Burner configurations not yet developed for LFR and GFR
  - Compare general performance based on physics data
- **Neutron energy spectra compared for the three concepts**
  - One-group cross sections generated in consistent manner
  - Overall neutron balance assessed using "D-factor" technique
    - *Quantifies neutron consumption for complete destruction*
- **Collaborative work with CEA to compute the D-factors with alternative techniques**
  - Excellent agreement observed
  - Technique applied to variable fuel/moderator LWRs by CEA

Comparison of Generation-IV Fast Reactor Spectra



- **Metal-fuel SFR exhibits the hardest spectrum**
- **GFR spectrum is softest because of SiC matrix moderation**
- **LFR spectrum depressed above 1 MeV from (n,2n) reactions**

Comparison of D-Factors for Key Actinide Isotopes

Isotope	Metal Fuel SFR	Oxide Fuel SFR	LFR	GFR	PWR (CEA)
U235	-1.036	-0.953	-0.918	-0.843	-0.385
U238	-0.898	-0.793	-0.711	-0.629	0.068
Np237	-0.878	-0.725	-0.648	-0.513	0.931
Pu239	-1.713	-1.609	-1.586	-1.446	-0.643
Pu240	-1.283	-1.143	-1.036	-0.943	0.557
Pu241	-1.419	-1.357	-1.287	-1.273	-0.370
Pu242	-1.126	-0.920	-0.724	-0.644	1.216
Am241	-0.929	-0.790	-0.670	-0.584	0.932

- **D-factor becomes more negative (indicating neutron surplus) as the spectrum hardens**
- **All FR options have favorable neutron balance, but significant variations (up to 30%) particularly for fertile species**
- **Fertile species are net consumers in thermal spectrum**
  - Thus, fissile enrichment required even at equilibrium

Extension of Tool for Comparing the Transmutation Potential of Fuel Cycles

- **D-factor comparison was extended to thermal reactors**
  - Wide range of spectral variation (fuel/moderator ratio)
  - Different fuel forms considered with unique cross section data
- **Recently, fuel cycle model has been extended**
  - Details in PHYSOR 2004 paper
  - Allows direct computation of equilibrium composition
  - Alternate fuel cycle strategies (e.g., once-through, recycle with losses, elemental management) can be modeled
- **Result is a powerful tool (based on basic physics data) for comparing diverse fuel cycle strategies**
  - Quick evaluation of mass flows and compositions
  - Key fuel cycle parameters (e.g., radiotoxicity, void coefficient, decay heat, etc.) derived from equilibrium concentration
  - Accuracy limited by one-group XS methods – inferior to modern fuel cycle analysis techniques, but useful for basic comparisons

Thermal Reactor Transmutation Analyses

LWR recycle assessment report (with CEA) was completed

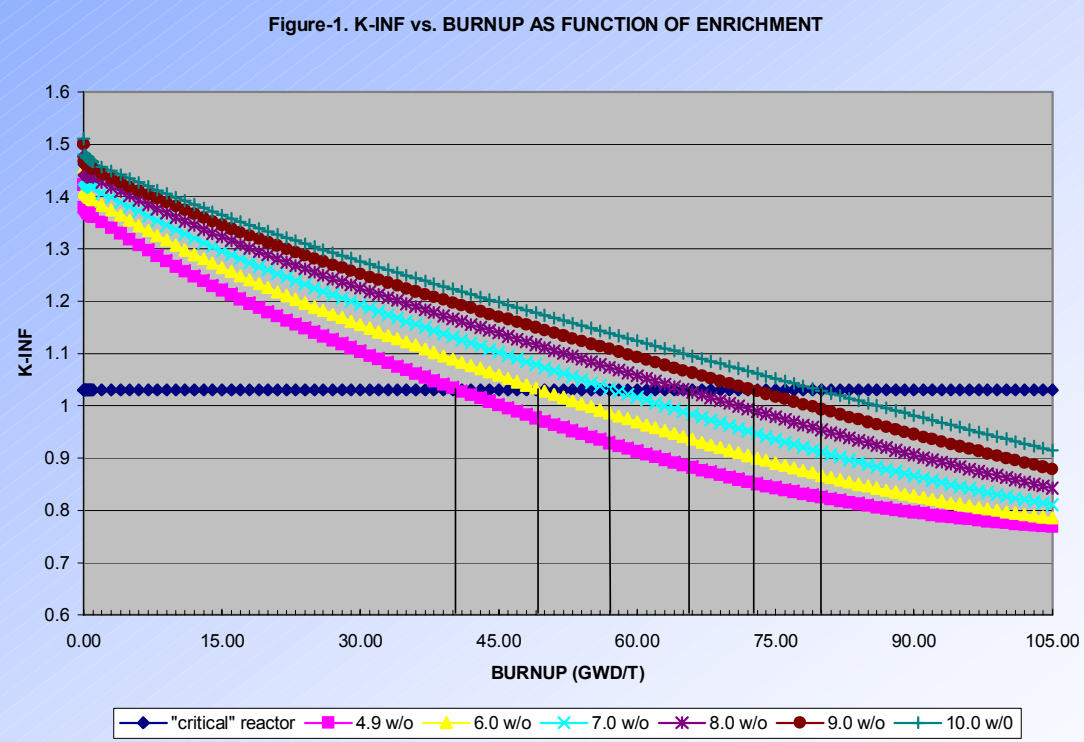
- **Review of international studies for heterogeneous Am recycle**
  - Concentrated targets placed in water holes
  - Minor actinide based pins
  - Mixed oxide targets, also in water holes
- **Work on inert matrix fuel (IMF) was summarized**
  - Most advanced proposal is solid solution in stabilized zirconia
- **The key transmutation features of other thermal concepts were also reviewed, with existing studies noted**
  - BWR performance compared to PWR
  - Gas-cooled thermal reactor options (e.g., HTGR, VHTR)
    - *Generally utilize TRISO fuel particles*
    - *Studies with non-uranium fuels for plutonium burning*
  - Reduced Moderation Water Reactor (RMWR)

HIGH-BURNUP FUELS

- Interest from utilities, and "recommended" in MIT Report to make nuclear more attractive
- Current NRC limit 62 GWD/T "average for the peak rod"
- NRC considering implications of "extended burnup" (75 GWD/T) and higher on
  - Cladding integrity & fuel design limits (resolved)
  - Control Rod Insertion Problems (resolved)
  - Criteria and analysis for reactivity accidents (active…likely to have lower limits)
  - Criteria and analysis for LOCA (active)
  - Criteria and analysis for BWR power oscillations/ATWS (active)
  - Fuel rod and neutronic computer codes for analysis (resolved)
  - Source term and core melt progression (resolved)
  - Transportation and dry storage (active)
  - High enrichments (>5%) (deferred)
- NEA/OECD Expert group will perform a technical assessment of potential of extending current LWR ceramic fuel burnup limits to approximately 100 GWD/MTU, and provide a written report in the next 2 years.
- Technical issues:
  - Power peaking due to high enrichment of fresh fuel
  - Boron coefficient and control rod worth decrease with increasing fuel enrichment
  - MTC increases "slightly" with fuel enrichment, but still negative
  - Fuel handling (higher heat, gamma, and neutron)
  - Thermal creep which is thought to be the dominant mechanism for rod failure with high burn-up fuel
  - Will new clad materials do the job?
  - Gas pressure (plenum, annular fuel?)
- Economics – recent papers from BNFL suggest "optimum" fuel cycle costs is around ~70 GWD/T (6-7 w/o)

K-inf vs. BURNUP, AS FUNCTION OF ENRICHMENT

- Standard 17x17 Westinghouse Assembly
- BOXER lattice physics code with JEF-based library
- Assume Δk=0.03 to account for core leakage
- Discharge burnup =
  - 60 GWD/T for UE of 4.9 w/o
  - 98 GWD/T for UE of 8.0 w/o



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PRODUCTION OF TRU vs. BURNUP

- With increased discharge burnup:
  - Total Pu reduced by 18%
  - Fissile Pu reduced by 25%
  - Total TRU reduced by 18%
  - Pu-238 fraction 2x
  - Non-Pu TRU increases by 16%

	BURNUP = 60-GWD/T		BURNUP= 98 GWD/T	
	Kg/Assembly Per GWD	Wt-%	Kg/Assembly Per GWD	Wt-%
Neptunium	7.95		8.70	
Plutonium	103		81.1	
Pu-238		3.51		6.99
Pu-239		48.94		44.67
Pu-240		25.01		23.79
Pu-241		14.59		14.93
Pu-242		7.95		9.61
Americium	2.60		3.00	
Curium	1.24		1.99	
TOTAL TRU	115		94.8	

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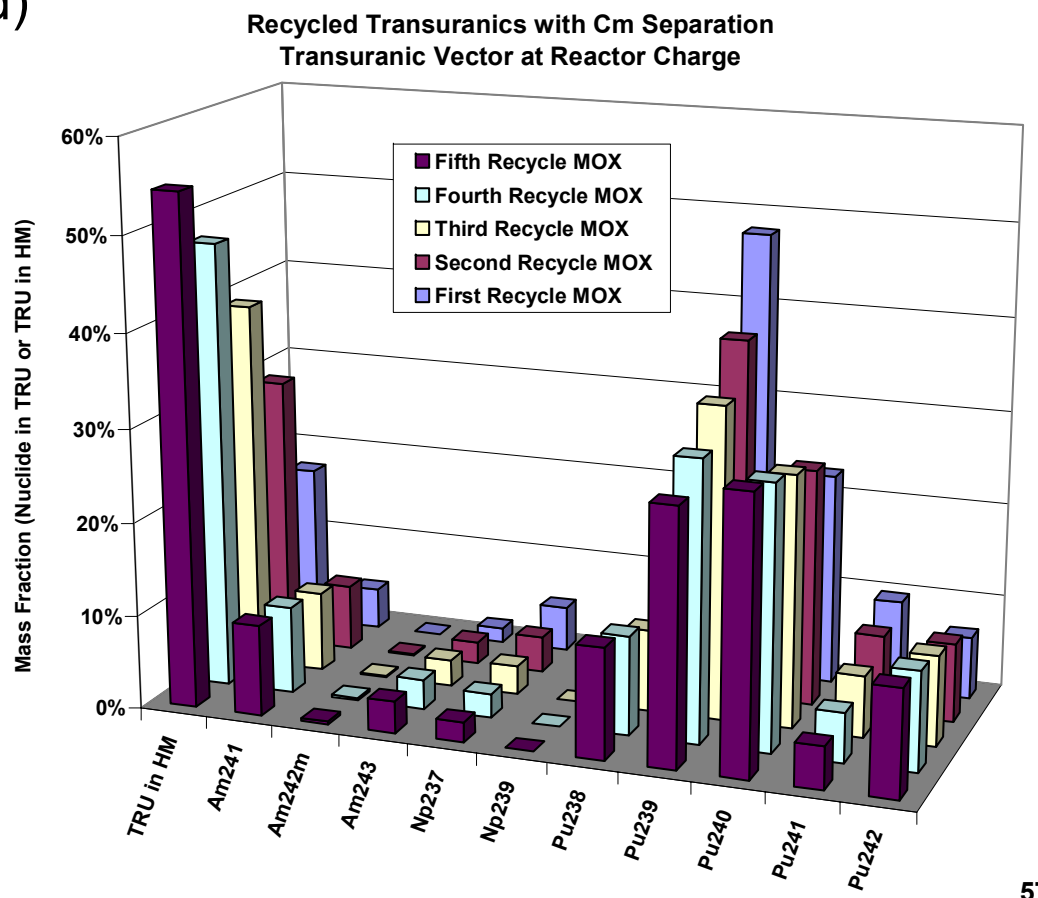
Homogenous Recycle of Am in LWRs

- **In current design, repository loading of spent nuclear fuel is thermally limited, particularly by long-term considerations**
- **To significantly reduce the long-term decay heat, plutonium (or Pu+Np) recycle is insufficient**
  - Am-241 is the dominant contributor
  - Short cooling time can limit Pu-241 decay, but may adversely impact the processing technology
  - Thus, Am recycle is desirable
- **Recycle of Pu+Np+Am was considers in three concepts**
  - Standard MOX
  - Heterogeneous UO2/MOX assembly (CORAIL concept)
  - Inert Matrix Fuel

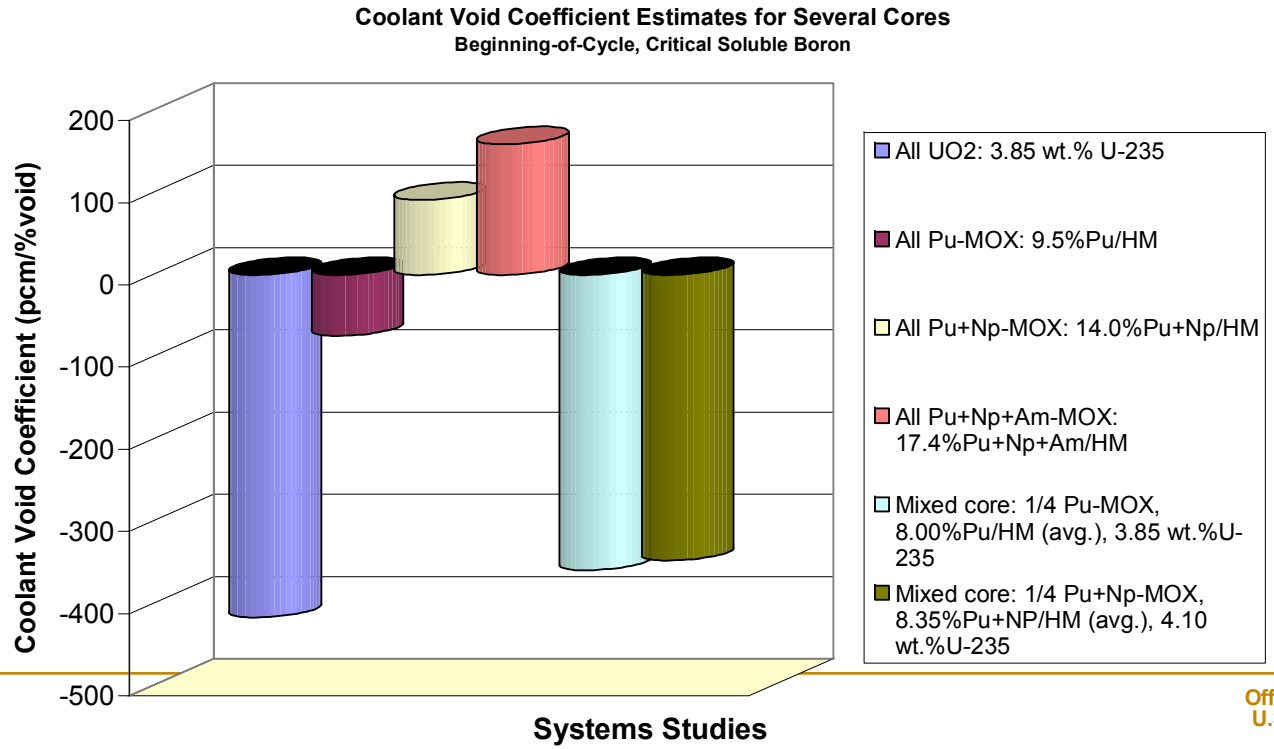
Limited Recycle in LWRs: Standard MOX Transmutation Results

- **Pu+Np+Am recycled in homogeneous MOX fuel assembly**
  - Initial source of transuranics harvested from spent UO_2 (51 GWd/MT, 5 years cooled)
  - Assembly-level calculations performed to MOX discharge burnup of 51 GWd/MT; TRU loading in MOX adjusted to meet discharge burnup constraint
  - Unconsumed Pu+Np+Am recycled
  - Fissile-TRU and Am-241-chain nuclides gradually consumed



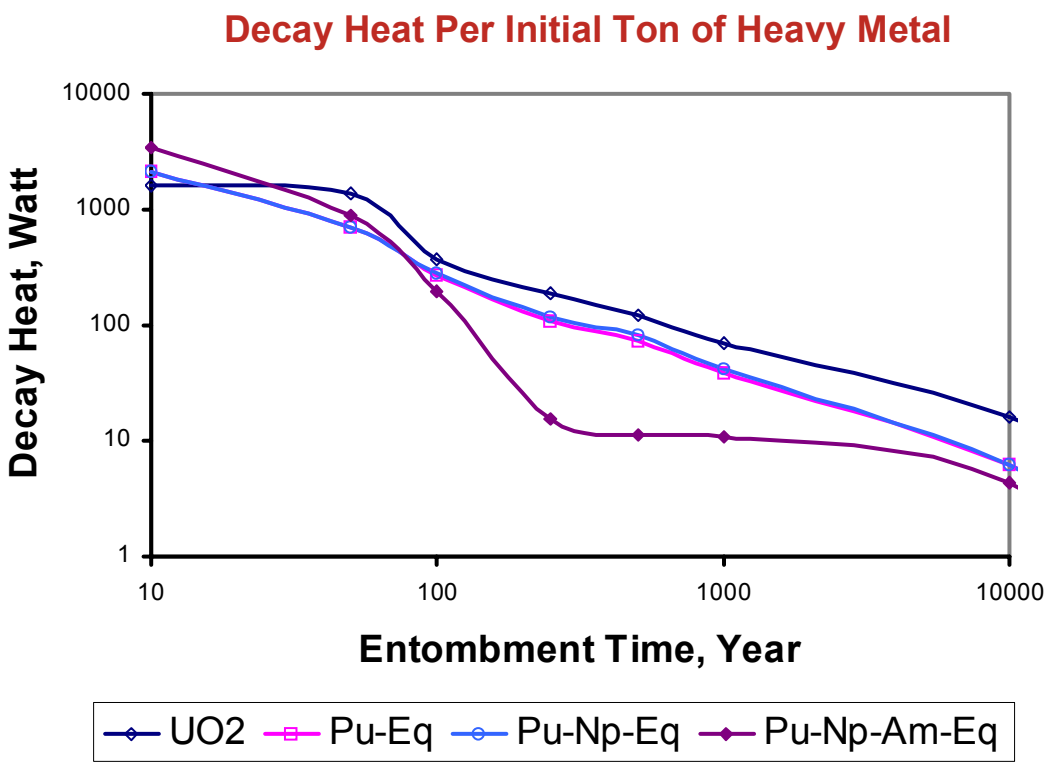
Limited Recycle in LWRs: Standard MOX Void Coefficient

- **Coolant void coefficient becomes more positive as TRU loading in MOX increases**
  - Full-core loading of Pu+Np+Am-MOX has large, positive void coefficient
  - Cores partially loaded with MOX have void coefficients only 15-20% less negative than all UO_2 core



Limited Recycle in LWRs: CORAIL Decay Heat Results

- **Am recycling reduces the decay heat relative to the Pu and Pu-Np cases**
  - With Am recycle, Cm-244 dominates decay heat from heavy metals in the short term
  - Presence of Cm-244 is source of Pu-240 which reduces relative benefit in the intermediate term (about 10,000 years)



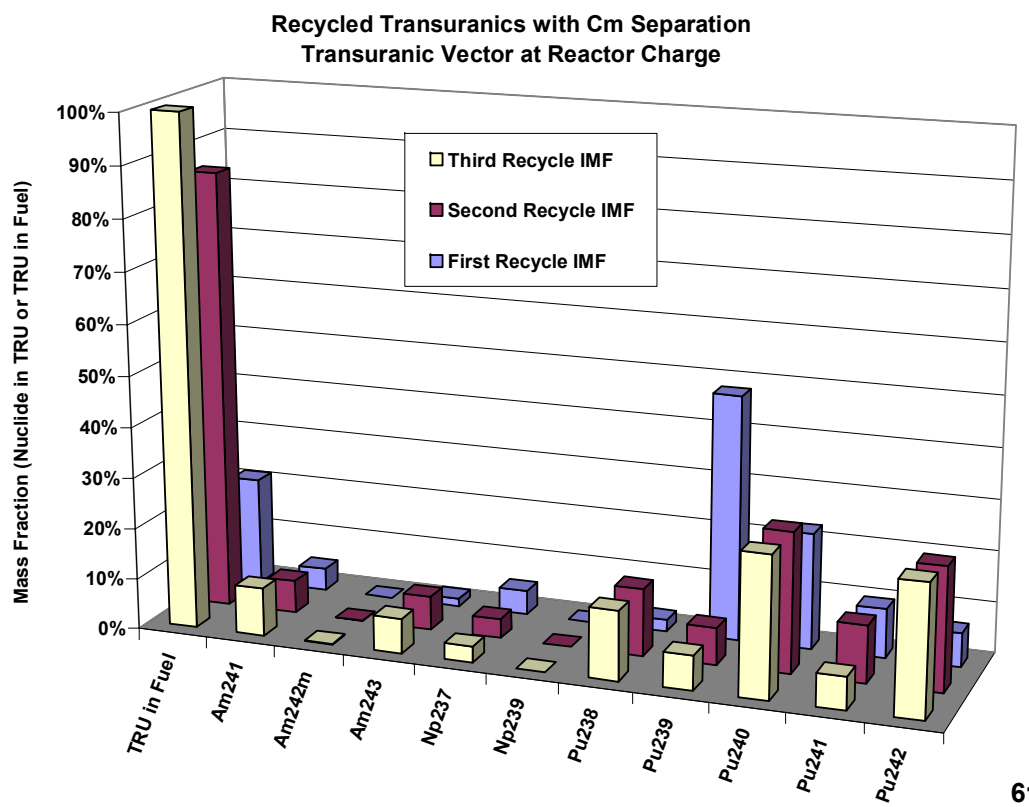
Am Recycle in CORAIL: Impact on Fuel Handling Metrics

		CORAIL-Pu (Reference)	CORAIL Pu+Np	CORAIL Pu+Np+Am
Decay Heat	Fabrication	1	1.9	5.9
	Charge	1	1.9	5.8
	Discharge	1	1	1
	After 5 years cooling	1	1.1	2.2
Neutron Source	Fabrication	1	1.5	3.8
	Charge	1	1.5	3.8
	Discharge	1	0.98	2.6
	After 5 years cooling	1	0.94	2.4
Gamma Source	Fabrication	1	1.9	13.7
	Charge	1	1.6	8.5
	Discharge	1	1	0.98
	After 5 years cooling	1	0.99	0.95

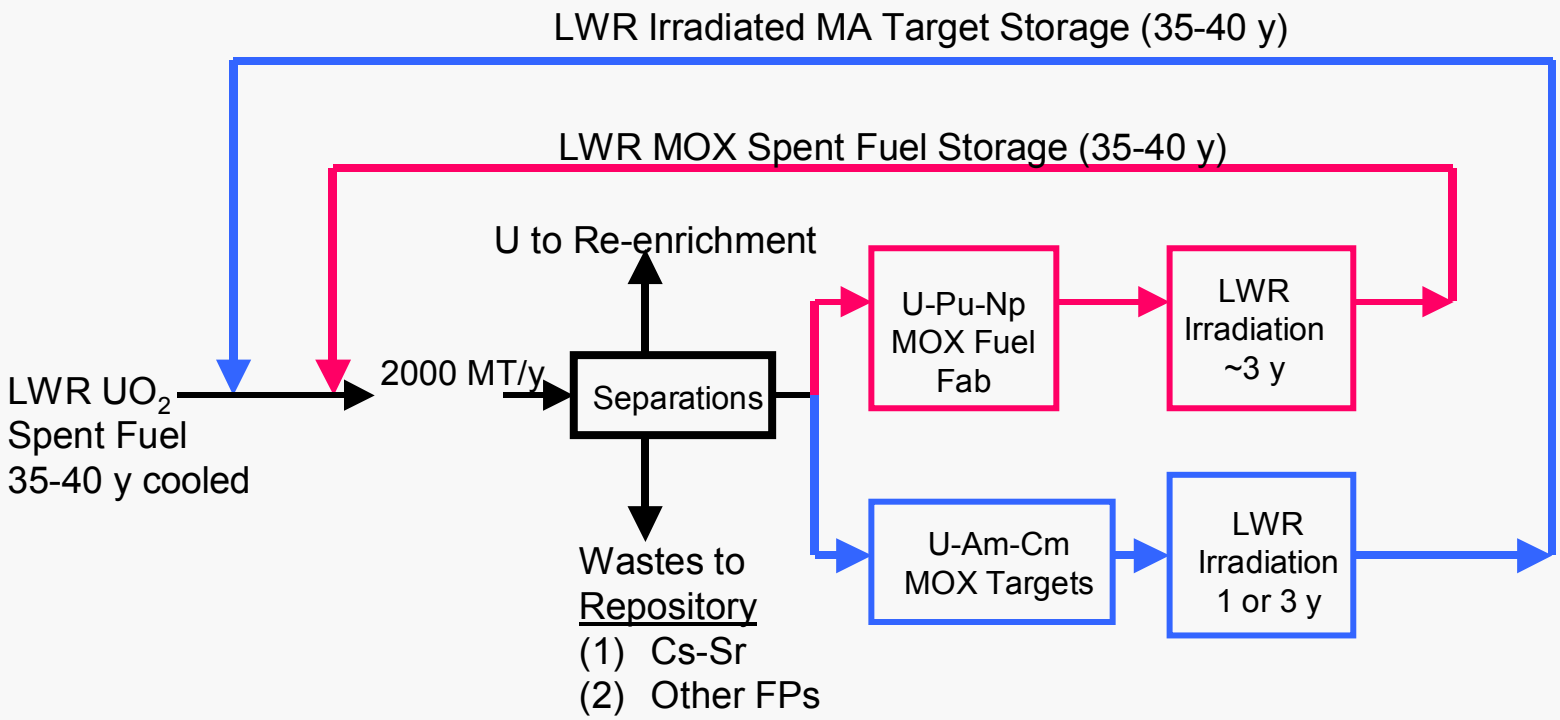
Am recycling increases decay heat and radiation sources during fuel handling operations

Limited Recycle in LWRs: IMF Transmutation Results

- **Inert-matrix fuels (IMF) eliminate production of transuranics from U-238 conversion**
  - Very useful in weapons-grade Pu disposition program
- **Evaluated use of solid-solution $TRUO_2$-ZrO_2 for Pu+Np+Am recycling**
  - Assembly-level calculations over typical PWR fuel assembly residence time
  - Significant consumption of fissile-TRU and Am-241-chain nuclides in first recycle alone
    - *94% of Pu-239 consumed*
    - *48% of Am-241-chain consumed*
  - Further recycles perhaps not practical



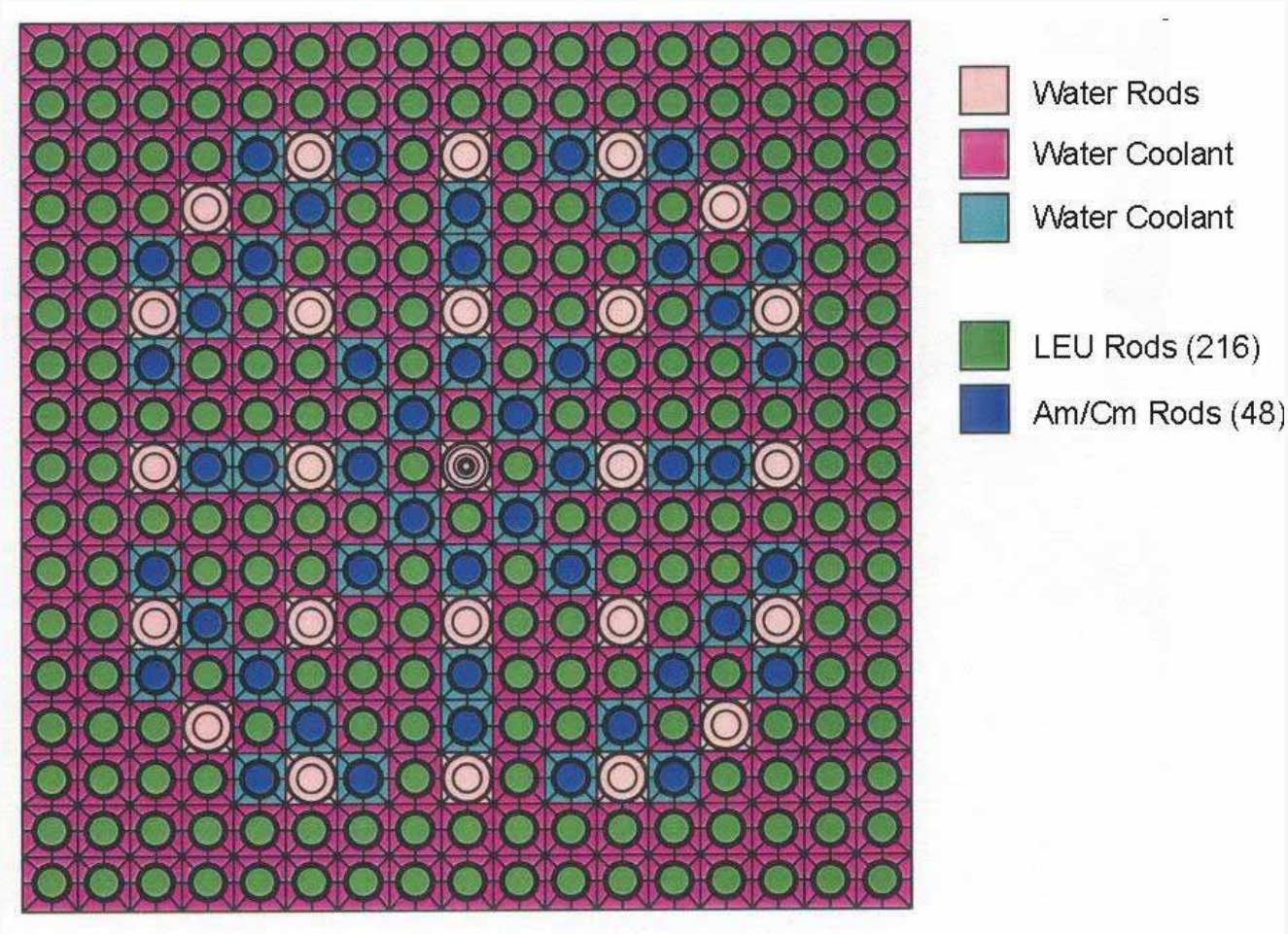
Am/Cm Transmutation Strategy



5-10 y Separations – Fuel Fab – Irradiation Period

1960s – 2015: LWR UO_2 Irradiations Only
2015 – 2055: LWR UO_2 + LWR MOX 1st Cycle Irradiations
2055 – 2095: LWR UO_2 + LWR MOX 2nd Cycle Irradiations

PWR Assembly Model for Initial Am/Cm Target Transmutation Study



Significant MA Reduction Can Be Achieved in LWRs Using MA Target Strategy

- **Assumptions:**
  - **2000 MT/yr processing of 35-40 yr-cooled LWR fuel**
  - **Am/Cm processed into pins w/4.5% LEU or inert diluent**
  - **Pu/Np processed into U-Pu-Np pins for 1/3 MOX cores**
  - **3yr irradiation w/3mo or 3yr cooldown between 18mo shuffle**
  - **35-40 yr cooldown (use oldest LWR SF first)**

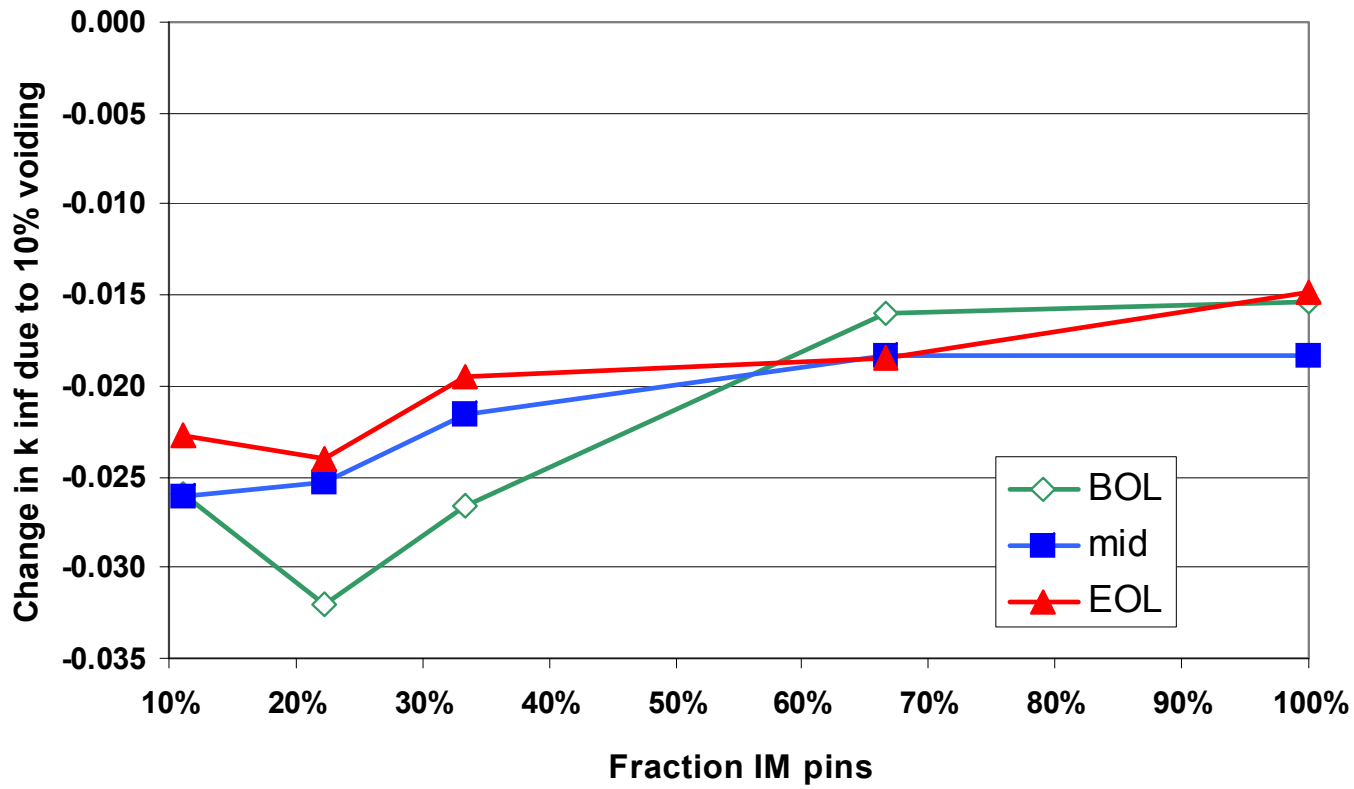
- **Results:**
  - **95% consumption of Am in target rods w/ inert diluent or 87% reduction for LEU diluent**
  - **At least 2 cycles possible before needing higher enrichment**
  - **Hence, keep MA out of repository for at least additional 75-80 yrs**

Trades Space for Am/Cm Management Task

- **Separations feed material:**
  - **30-year-old 4.5 wt-% LEU with 45 GWd/MTU burnup**
  - **5-year-old 4.5 wt-% LEU with 50-60 GWd/MTU burnup**
- **Target assembly options:**
  - **Number and arrangement of target rods in assembly**
  - **Minor actinide loading**
  - **Diluent (DU, NU, RU, LEU, MOX, IMF)**
  - **Driver fuel (LEU, MOX)**
- **Fuel cycle options:**
  - **Length (18 months)**
  - **Number of cycles (2, 3, several)**
  - **Delay before reload (1 month, 3 years)**

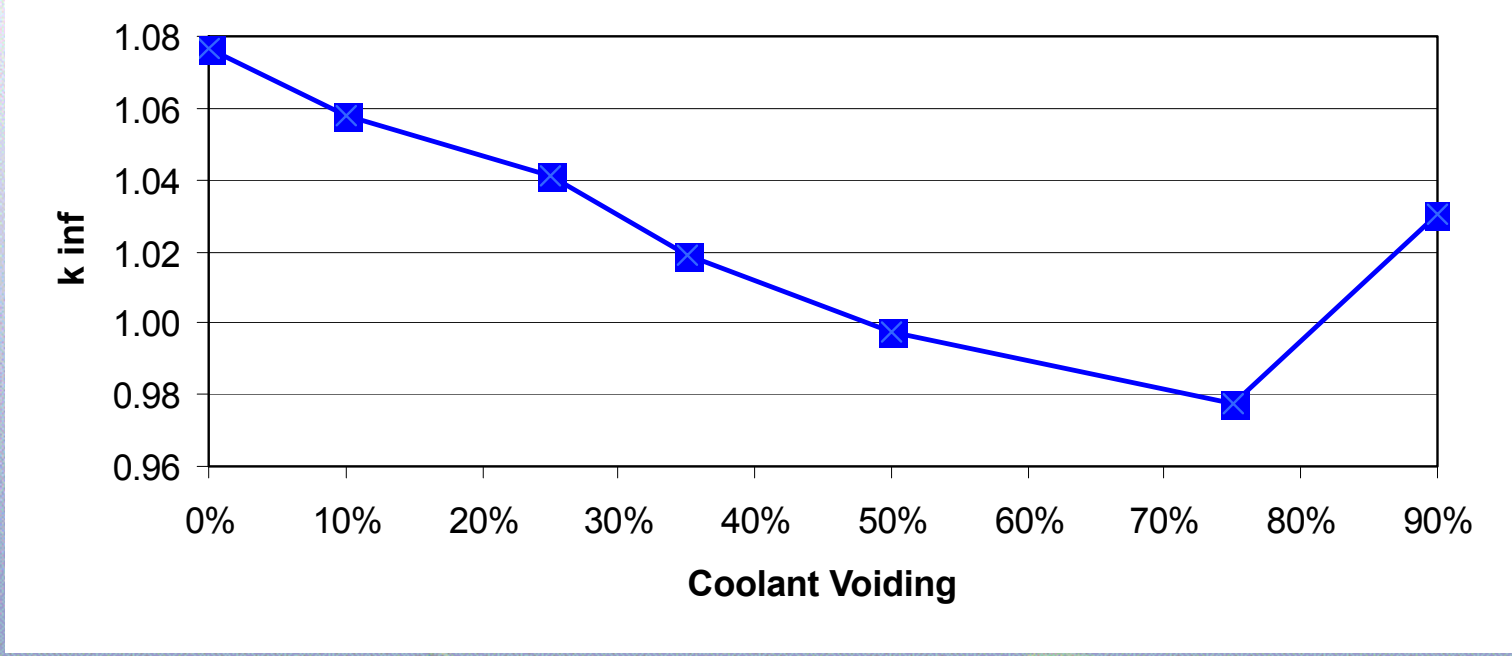
Preliminary calculations of voiding coefficients for arrays of UO_2 and IMF rods

- IMF rods contained 8 atom % Pu+MA in yttria-zirconia matrix in typical 17x17 configuration
- Coolant density reduced 10%
- Sufficient histories run until std dev was 0.1 x change in k inf from full coolant density
- Isotopic content at BOL, mid-life (720 full power days) and EOL (1440 fpd) used



Changes in kinf due to coolant voiding in an array of IMF rods (no UO_2)

- **17x17 configuration, all IMF rods, mid-life (720 fpd)**
- **Coolant density reduced 10%, 25%, 35%, 50%, 75% and 90%**
- **Sufficient histories run until std dev was about 0.0025 (same criterion as previously)**



Fast Reactor Transmutation Analyses

Fast reactors with closed fuel cycle can effectively manage TRU

- **Can be configured as modest breeders (CR≥1) to moderate burners (CR≥0.5) with conventional technology**
- **Low conversion ratio designs (CR<0.5) have been investigated for transmutation applications in AFCI**
  - High enrichment fuels are requires (~50% TRU/HM for CR=0.25)
  - Non-uranium fuel would be needed to achieve CR=0
- **Safety performance will change at low uranium content (e.g., reactivity losses, reduced Doppler coefficient)**
  - Detailed safety analysis conducted for CR=0.25 SFR system
  - Passive safety behavior is not compromised
- **Work has been initiated to develop fast burner designs for the other Generation-IV concepts**
  - Preliminary GFR results will be shown

Low Conversion Ratio SFR Transmutation Performance

	High Leakage			Compact Core		
Targeted CR	**0.50**	**0.25**	**0.0**	**0.50**	**0.25**	**0.0**
Fuel Pin Diameter (cm)	0.74	0.64	0.49	0.67	0.59	0.45
Fuel Volume Fraction	0.38	0.26	0.14	0.30	0.22	0.12
Avg. Fuel Enrichment (%)	**30**	**51**	**90**	**31**	**51**	**91**
Heavy Metal Loading (MT)	14	8.4	4.1	7.5	4.6	2.5
TRU Loading (MT)	4.2	4.1	4.1	2.3	2.3	2.5
External Feed (MT/yr)	2.2	1.3	0.72	2.2	1.4	0.85
TRU consumption rate (kg/yr)	**126**	**198**	**270**	**122**	**193**	**259**
Burnup Reactivity Loss (%Δk)	2.8	4.6	6.4	2.6	4.3	5.9

- **Low conversion ratio designs developed for both high leakage (D=3.8 m) and compact (D=1.9 m) configurations**
- **Conventional enrichment at CR ~ 0.5**
- **Gradually increases to roughly 50% TRU/HM at CR ~ 0.25**
  - Zr content also steadily increased up to 40% for nonuranium case
- **TRU Production Rate of LWR is ~250 kg/yr for 1 GWe**

Safety Analysis of Low Conversion Ratio SFR

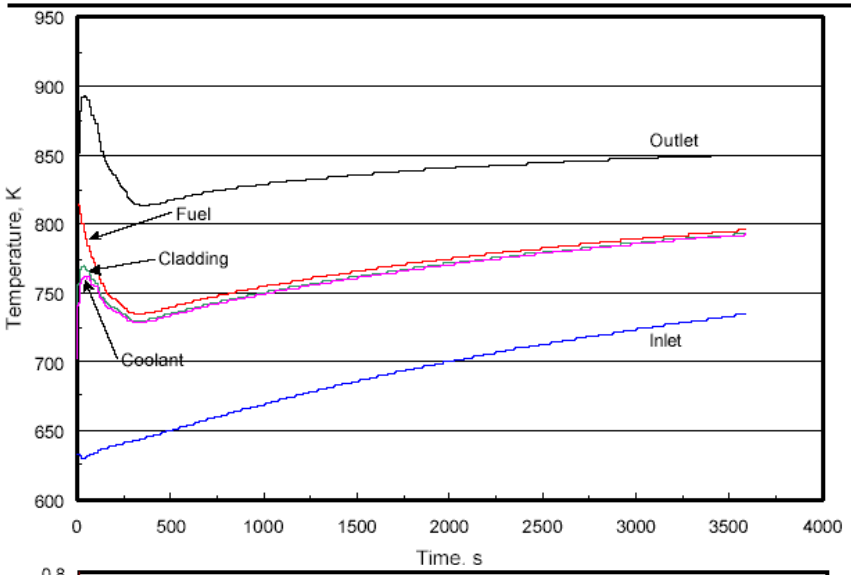
Three double fault accident scenarios were considered with simplified safety model

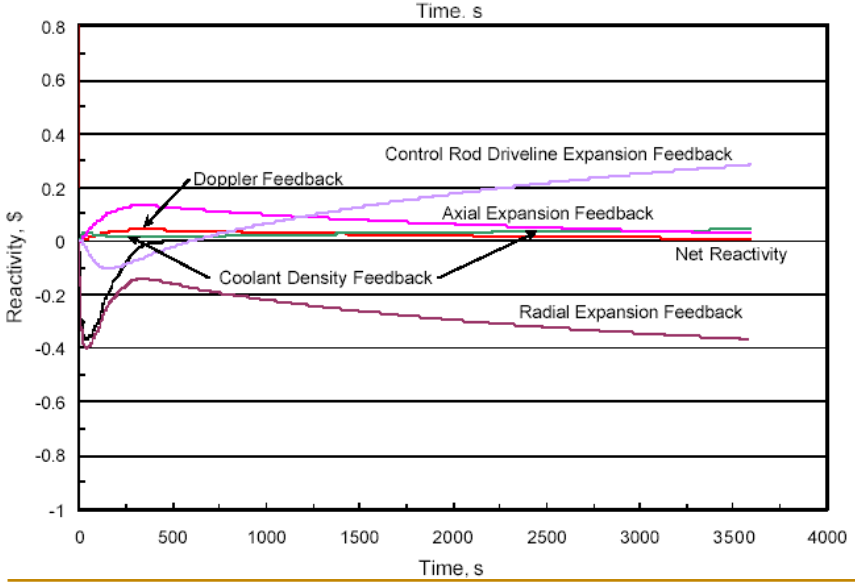
- **Loss of flow**
- **Transient overpower**
- **Loss of heat sink**

Detailed SAS4A model developed for CR=0.25 core and loss of flow event investigated in detail

- **High power density and reactivity loss**
- **Still very high level of self-protection**
- **Boiling/Damage margins similar to simplified model results**
- **Sensitivity to modeling/coefficient uncertainties evaluated**
- **Details in PHYSOR 2004 paper and AFCI Report**

Safety Analysis Example: Unprotected Loss of Flow Transient





- **Both primary and secondary Coolant pumps are assumed to lose power.**
  - Heat removal through the steam generator decreases as coolant flow decreases.
- **Failure of the reactor scram system.**

<u>General Behavior</u>

Flow decreases.

Fuel/Coolant temps. Increase.

Expansion coefficients drive the reactivity negative.

Power decreases.

Safety Analysis Results: Temperature Variations for Loss of Flow Event

	High Leakage			Compact Core		
Conversion Ratio	0.50	0.25	0.00	0.50	0.25	0.00
Peak Avg. Fuel Temp.	804	812	828	863	868	885
Peak Coolant Outlet Temp.	837	816	794	962	948	931
Peak Avg. Core Coolant Temp.	753	749	744	825	821	815

- All temperatures (K) are within safe operating limits.

  Coolant: ~1250 K Fuel Melts: >1300 K Clad Melts: ~1700 K
- Temperatures relatively insensitive to the conversion ratio.
- Temperatures higher for compact core, in part, because of absence of GEMs.
- Temperature Transients are small

Steady State	Avg. Cool	Avg. Fuel
High Leakage	703	800
Compact	705	850

Design Approach for Low Conversion Ratio GFR

- **Gas-cooled fast reactor (GFR) burner designs to achieve low conversion ratio have also been explored**
- **Two GFR design approaches were considered**
  - Block Fuel
    - *Carbide particle fuel*
    - *Embedded in SiC matrix/structure*
    - *Fuel volume fraction reduced by Fuel / Matrix ratio*
  - Pin Fuel
    - *Solid pin carbide fuel*
    - *Nb-1Zr structure and cladding*
    - *Similar approach to SFR, fuel volume fraction reduced by Pin Diameter (modify Fuel/Coolant ratio)*

Low Conversion Ratio GFR Transmutation Performance

	Block				Pin			
Conversion Ratio	0.79	0.52	0.24	0.00	0.57	0.53	0.26	0.00
Volume Fraction								
Fuel	0.24	0.17	0.10	0.06	0.29	0.27	0.16	0.07
Structure/Matrix	0.20	0.27	0.34	0.38	0.16	0.15	0.11	0.08
Coolant	0.56	0.56	0.56	0.56	0.53	0.57	0.72	0.84
TRU Enrichment (%)	23	33	57	100	25	27	44	100
TRU consumption rate (kg/yr)	41	99	167	228	86	96	157	229
Burnup reactivity loss (%Δk)	1.5	3.6	6.1	8.6	2.8	3.2	5.6	8.2

- Again biggest penalty is sharp increase of burnup reactivity loss
- Reduced fuel volume fraction may be advantageous for block design with particle fuel – more reasonable particle fraction
- Safety coefficient trends similar to deep burner SFR results