

## **Nuclear Proliferation Risk Mitigation Approaches and Impacts in the Recycle of Used Nuclear Fuel in the USA**

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### **ABSTRACT**

EnergySolutions and its team partners, which include the UK National Nuclear Laboratory (NNL), are one of four industry teams to have received an award from the US Department of Energy to carry out design studies in support of the US Global Nuclear Energy Partnership (GNEP). This team has developed a detailed scenario model for a future US nuclear fuel cycle based on a closed used nuclear fuel recycle as an alternative to the current once-through-and-store system. This scenario enables the uranium and plutonium in Light Water Reactor (LWR) used fuel from the current reactor fleet, and from a fleet of replacement LWRs, to be recycled as both Uranium Oxide and Mixed Oxide (MOX) fuel using reprocessing plants that conform to the requirements of GNEP. There is also a provision for “burning” in thermal reactors certain long-lived transuranics (Np, Am, Cm) formed into targets. The residual fission product waste, without these long-term heat emitters, will be vitrified and consigned to the US National Geologic repository. Later in the scenario a fleet of Advanced Recycle Reactors (ARR), based on sodium cooled fast reactor technology, are introduced to enable full transmutation of all transuranics and thus attain the GNEP sustainability goal. The recycle scenario avoids the need for the Yucca Mountain repository to receive unprocessed used nuclear fuel and is effective at prolonging its lifetime and delaying the need for a second repository. This paper explains the process by which EnergySolutions selected the U-Pu and U-Pu-Np MOX products and the technological requirements for the recycle plants and describes materials flow analysis that has been carried for the US nuclear fuel cycle scenario using NNL’s ORION scenario modelling program [1]. One of the prime requisites of GNEP is to ensure that the risk of proliferation is minimized and the paper describes NNL’s approach to objectively assessing the proliferation risk of the scenario relative to that of a conventional recycle scheme.

### **INTRODUCTION**

EnergySolutions and its team partners (the ES Team), which include the UK National Nuclear Laboratory (NNL), are one of four industry teams to have received an award from the US Department of Energy (DOE) to carry out design studies in support of the US Global Nuclear Energy Partnership (GNEP). This team has developed a detailed scenario model for a future US nuclear fuel cycle based on a closed used nuclear fuel (UNF) recycle as an alternative to the current once-through-and-store system. This scenario enables the uranium and plutonium in Light Water Reactor (LWR) UNF from the current reactor fleet, and from a fleet of replacement LWRs, to be recycled as both Uranium Oxide and Mixed Oxide (MOX) fuel using reprocessing plants that conform to the requirements of GNEP. There is also a provision for “burning” in thermal reactors certain transuranics (Np, Am, Cm) formed into fuel and targets. The residual fission product waste, without these long-term heat emitters, will be vitrified and consigned to the US National Geologic repository. Later in the scenario a fleet of Advanced Recycle Reactors (ARR), based on sodium cooled fast reactor technology, are introduced to enable full transmutation of all transuranics and thus attain full sustainability. The recycle scenario avoids the need for the Yucca

Mountain repository to receive UNF and is effective at prolonging its lifetime and delaying the need for a second repository.

One of the fundamental tenets of GNEP is to maximise long term sustainability of the nuclear fuel cycle by recycling fissile materials, but with the stipulation that the recycle technology should demonstrate inherent proliferation resistance. Conventional PUREX reprocessing technology as used already in France, Japan, Russia and the UK involves the separation of PuO<sub>2</sub> powder that has only traces of fission products and other transuranics (apart from Am-241 in-growth from Pu-241 decay). The PUREX process is established in commercial-scale plants in all these countries. There are no technological unknowns associated with PUREX and it is also proven to be economically viable and competitive as a used fuel management option.

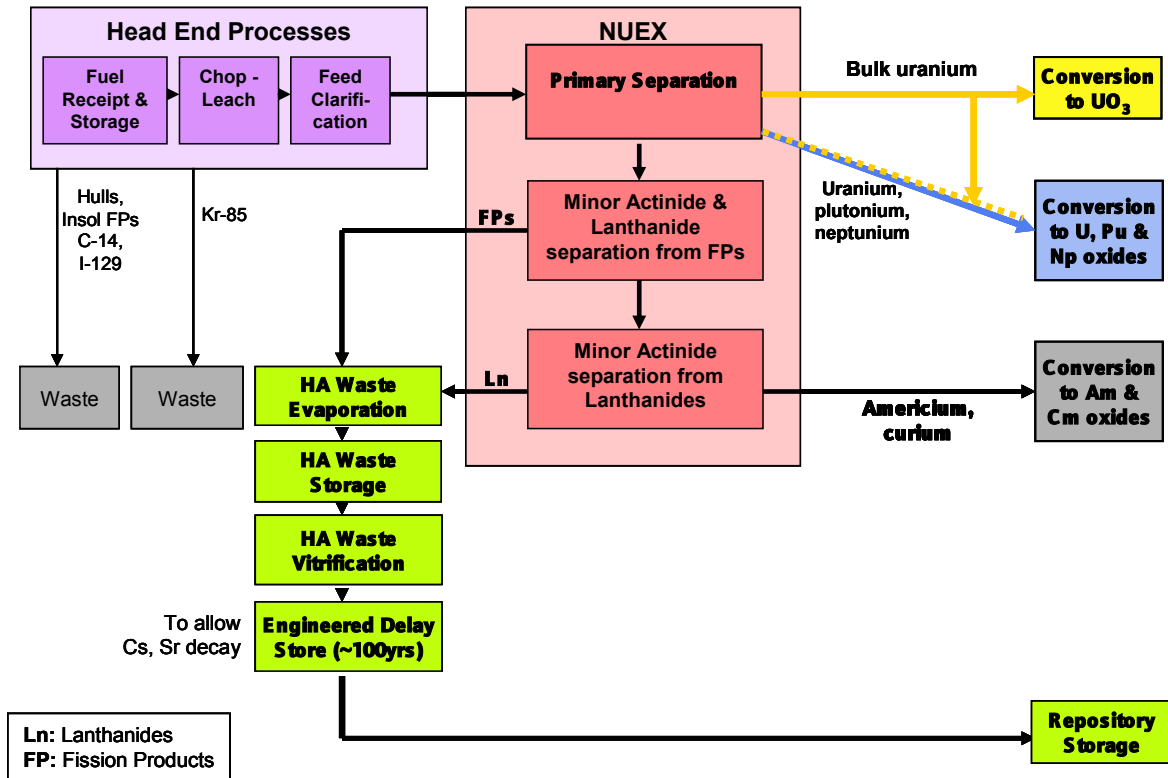
The principal difficulty with the PUREX process in the context of GNEP, is its separation of pure PuO<sub>2</sub> powder, that is potentially useable for nuclear weapons and is also accessible to the state in which the plant operates. It must be stressed that plutonium from current Light Water Reactor (LWR) operations, typically with average burnups approaching 50 GWd/t, is far from ideal for weapons usage, since it has a strong intrinsic neutron source from Pu-240 and a significant decay heat output from Pu-238 decay, both of which factors would severely complicate its use in a weapon. Nevertheless, it is widely accepted that such plutonium is potentially usable in an inefficient fission device. Moreover, such plutonium would certainly pose a risk following unauthorized theft or diversion, for use either as a low-yield fissile device or as a dispersal terror weapon and this is the reason why stringent security precautions are in place at reprocessing plants and their associated plutonium stores. Best practice is that the security risk is minimized by promptly re-using the plutonium for MOX fuel manufacture.

The ES Team's overall GNEP scheme resolves this proliferation issue by adapting the reprocessing separation processes so that no pure plutonium is separated anywhere in the process. Instead, the products from this "NUEX" separation are a bulk uranium stream and a mixed plutonium-uranium stream. The NUEX flowsheet was initially developed to provide U:Pu mass ratios in the range 1:1 to 3:1 but higher ratios can be readily obtained and are being examined. This stream also, optionally, contains the neptunium. The uranium and uranium-plutonium product streams are converted to oxides ready for recycle to thermal reactors as UOX and MOX fuel respectively. The use of Atomic Energy of Canada's (AECL's) CANDU reactors for recycle of the UOX product is part of the ES Team's scheme because LWR recycled uranium does not need re-enrichment for this purpose. AECL is a member of the ES Team. The ES Team's overall GNEP scheme has been modeled to explore all such future fuel cycle options and provide cost estimates for them. It is underpinned by detailed materials flow calculations that have been obtained from the NNL ORION scenario program and the results from ORION have also fed into a proliferation risk assessment, which this paper summarizes.

## **SELECTION OF NUEX U-Pu-Np CO-PROCESSING**

A prerequisite of GNEP is that recycle plants should seek to minimize the risks of state diversion of PuO<sub>2</sub> and the terrorist threat, preferably through intrinsic measures rather than institutional controls. A uranium-plutonium co-product with neptunium as an optional addition was chosen by Energy Solutions as the basis of its GNEP submissions. In this NUEX process (Figure 1), the used nuclear fuel is dissolved in nitric acid and then the uranium, plutonium and neptunium are co-extracted into a conventional solvent of tri-butyl phosphate dissolved in a diluent such as "odorless" kerosene (TBP/OK) in the first stage of primary separation. In the following Strip stage, all the plutonium and neptunium, plus part of the uranium, are co-stripped from the solvent into a new aqueous solution. The amount of uranium that is co-stripped is readily altered by adjusting the redox and strip chemistry so that, in principle, the required U to Pu ratio for the MOX fuel can be produced. There is also the ability to blend further uranium into the mix downstream, or at the oxide powder stage, to obtain precise blend compositions and to make use of

depleted uranium stocks, if this is required. Americium and curium (Am & Cm) are extracted from the primary separation high level waste stream by separate processes and sent for target manufacture ready for transmutation in thermal or fast reactors.



**Figure 1: The NUEX Separation Process for Used Nuclear Fuel**

NUEX primary separation thus produces a bulk uranium stream plus a mixed uranium-plutonium-neptunium stream. Both streams are then purified in separate secondary stages, with the bulk uranium being mainly purified from technetium and the mixed stream being mainly purified from ruthenium. The mixed stream purification stage also provides the ability to remove neptunium, if this is not required in the MOX fuel. The mixed uranium-plutonium-neptunium product is beneficial for non-proliferation and security, with several positive attributes:

1. The presence of uranium would necessitate a chemical separation step to recover PuO<sub>2</sub>, which makes the plutonium less immediately accessible for mis-use by the state and potentially increases detectability of mis-use through routine international safeguards measures.
2. The uranium bulks up the volume of the co-product, increasing the mass that a terrorist organization would need to obtain.
3. When irradiated in recycled fuel, the Np-237 generates Pu-238 through neutron captures, significantly enhancing the Pu-238 inventory at discharge and adding to the intrinsic heat source of the plutonium. If recycled a second time, or even if it is retained in used fuel, this considerably diminishes the attractiveness of the plutonium for weapons use. The Pu-238, which is non-fissile, is particularly effective as an inherent barrier because it would require isotopic separation techniques to remove it rather than chemical ones.
4. Recycle of Np-237, instead of consigning it to the vitrified HL waste stream, helps reduce the long term radiotoxicity of the HL waste dispositioned in the geological repository.

The NUEX separation process is a chemical flowsheet modification to the modern conventional reprocessing processes already in successful commercial use in the UK and France, and soon to go into routine operation in Japan. Therefore all the process equipment is well proven and does not need further development. All that is required to bring NUEX into successful commercial operation is the final fully integrated demonstration of the chemical flowsheet. This is planned to be done at small scale (typically 1/5000 on flows) initially using simulant process streams “spiked” with key radioisotopes and then by similar small scale flowsheet demonstration in hot cells using actual UNF. Because of the significant development work, analysis and modeling of similar processes already carried out, this will be a low risk development program. Therefore, the ES Team estimates that a recycling plant using NUEX separation technology could be in commercial use by 2023, if design work started in 2009.

## **MODELLING OF THE US REACTOR FLEET AND ITS FUEL CYCLE WITH THE ORION PROGRAM**

### ***ORION Overview***

Working in close collaboration with the ES Team, NNL modeled a detailed future scenario for the entire US reactor fleet using its scenario modeling program ORION. Starting with the existing US reactor fleet as it was in 2007 and with a cache of used fuel closely representing that accumulated from historic reactor operations, ORION models the future evolution of the reactor fleet under the particular scenario assumptions agreed with the ES Team.

ORION (see Figure 2) is Windows program that was developed by NNL to calculate the material flows from a network of nuclear reactors and associated fuel cycle plants. ORION starts with uranium ore and calculates how the uranium is transmuted by fission and other nuclear reactions to yield the UNF compositions. The UNF can subsequently be specified as being routed to repository, in which case the decay of the nuclear inventory is tracked, or it can be specified to be recycled, in which case partitioned streams of different products are produced, some of which (e.g. fission product waste) is eventually disposed of in a repository and some of which (e.g. uranium & plutonium) can be recycled. ORION incorporates within it methods which allow it to calculate the isotopic evolution of nuclear fuel during irradiation. It also automatically tracks radioactive decay at each stage, taking account of decay transformations between nuclides. ORION is an invaluable tool because it allows the time evolution of a realistic reactor scenario to be modeled. Previously, it has often been the case that future reactor scenarios have been modeled on the basis of idealized scenarios in which different reactor fleets are assumed to be operating in an equilibrium condition. While it can yield some valuable insights, the difficulty with this equilibrium approach is that it is often not practically attainable in the finite timescales of realistic scenarios. For example, true equilibrium operation of a fleet of thermal and fast reactors is often not attained even after hundreds of years. Non-equilibrium effects, such as the eventual need to dispose of nuclide inventories that are being actively recycled at the point at which the reactor fleet is assumed to be phased out, can seriously affect the conclusions reached. This was in part the motivation for the ES Team and NNL to model the US reactor fleet in detail, as it provides a sound context in which to evaluate the performance of the favored U-Pu-Np recycle scheme.

Figure 2 shows a screen capture of the ORION program, showing all the interconnections in the US reactor scenario. Each node represents a different reactor type (with individual reactors of different types conglomerated under a single node), interconnected with each other via various fuel cycle nodes (which can represent passive plants such as stores, buffers or cooling ponds, or active plants such as recycle and fuel fabrication facilities). The arrows indicate the interconnections and direction of material flows between the different nodes. The scenario tracks material flows from 2007 onwards, with uranium ore (represented by the buffer in the top left) as the starting point for the materials flows and geological

repositories as the ultimate end point. All the node properties and interconnections are defined by the user using definitions tables that are invoked by clicking on the corresponding icon. In principle, ORION can track the entire nuclide chain, though in practice a reduced nuclide set (approximately 100 nuclides) is used to provide a practical run time and to limit the data storage requirements.

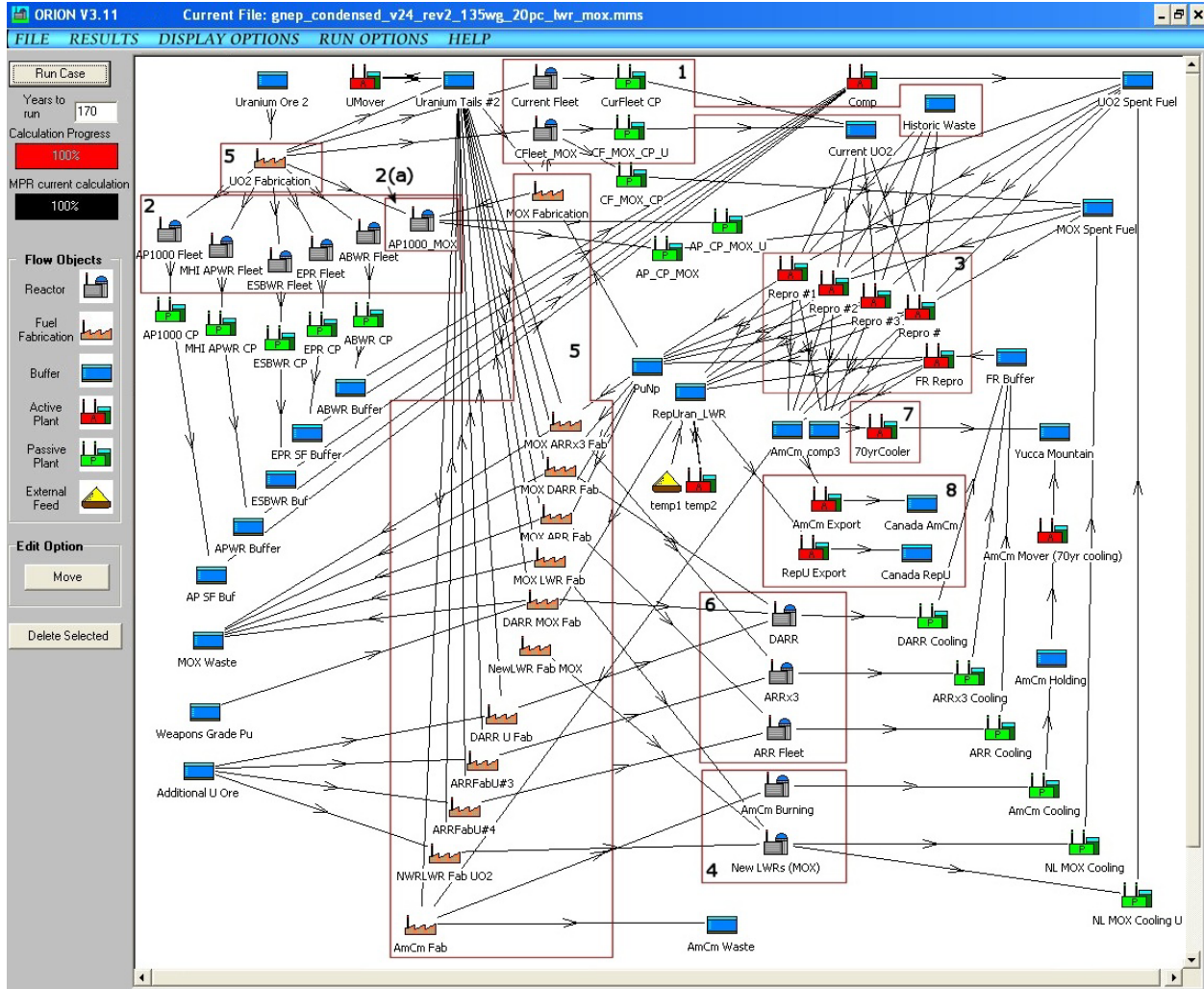


Figure 2: The ORION model of the GNEP scenario

### ORION Scenario

Although there are some aspects of the ORION scenario model that are simplifications, it is nevertheless very complex. This section describes the scenario model at a level of detail which is sufficient to establish the context for this paper, but nevertheless not in complete detail. The various nodes in Figure 2 have been grouped into numbered sections. This grouping helps with understanding the scenario and provides a logical framework for the discussion:

#### Section 1 – historic used fuel and future used fuel commitment from the current nuclear fleet

Section 1 represents the current US reactor fleet comprising the 104 commercial plants that are currently operational and the 55000 tons heavy metal (tHM) stock of historic UNF that was accumulated up to 2002 from these plants and the other plants that are now permanently shutdown or decommissioned. The

start date for the ORION model was taken to be 2007 and the estimated used fuel arisings from 2003 to 2007 were added to the 55000 tHM. The historic UNF is stored in a buffer (this is positioned outside and to the right of the main part of Section 1). The fuel cycle inventory code FISPIN [2] was used to estimate the nuclides inventories in the historic UNF, all decayed to 2007, and the data transferred to the ORION buffer. This approach decouples future operations from historic operations and simplifies the ORION model. The ORION model includes provision for the current plants to operate with up to one-third MOX core loading at some future data as required in the scenario once the recycle plants are operational. These current plants are assigned priority for MOX utilization.

*Section 2 – 1<sup>st</sup> generation (“COL”) new build LWR fleet*

This section represents a new fleet of Combined Operating and Licensing (COL) plants that are assumed be built and commissioned between 2008 and 2026, with the first becoming operational in 2017. The COL fleet comprises a total of 31 plants with a combined capacity of 40.75 GWe. The COL fleet is assumed to be a mix of different reactor designs from different vendors, Toshiba/Westinghouse AP-1000, MHI APWR, Hitachi/General Electric ABWR and ESBWR and AREVA US-EPR. Table 1 gives the design parameters assumed in the model for the different plant designs:

**Table 1 : Design parameters for the 1<sup>st</sup> generation new build (COL plants).**

| Parameter                  | APWR     | ABWR    | US-EPR  | AP1000 | ESBWR   |
|----------------------------|----------|---------|---------|--------|---------|
| Thermal Power (MWth)       | 4451     | 3926    | 4500    | 3400   | 4500    |
| Electrical Power (MWe)     | 1700     | 1356    | 1600    | 1117   | 1560    |
| Load Factor (%)            | 93%      | 87%     | 93%     | 93%    | 93%     |
| Core Inventory (tHM)       | 136.569  | 140.663 | 127.128 | 84.638 | 162.928 |
| Reactor Life (years)       | 60 years |         |         |        |         |
| Fuel Dwell Time (years)    | 5        | 5       | 5       | 4      | 5       |
| Discharge Burnup (GWd/tHM) | 55.4     | 44.3    | 60.1    | 54.6   | 46.9    |
| U-235 Enrichment (w/o)     | 4.8      | 4.4     | 4.8     | 4.8    | 4.4     |

After 5 years of cooling, the UO<sub>2</sub> UNF from the COL plants is transferred via intermediate buffers to a combined buffer titled “UO<sub>2</sub> used fuel”. The material remains in this buffer until it is required for recycle.

*Section 2(a) – 1<sup>st</sup> generation (“COL”) new build LWR fleet – MOX fuelled reactors*

There was a requirement to minimize the amount of separated fissile material. In order to achieve this goal, a fraction of the “COL” reactor fleet is allowed to load MOX fuel. For simplicity, the AP1000 plants were chosen since these reactors equate to approximately half the total capacity of the “COL” reactor fleet. Each AP1000 is limited to a 33% MOX load. If there is insufficient plutonium in any one year for MOX fuel, the shortfall is made up using additional UO<sub>2</sub> fuel.

*Section 3 – Used nuclear fuel recycle*

The ORION model assumes that four recycle plants start operating in 2025, 2035, 2065 and 2075 respectively. The first reprocessing plant is assumed to have a yearly throughput of 1500 tHM/yr whereas the remaining three plants have a higher throughput of 3000 tHM/yr. All reprocessing plants assume the same separation factors listed below. Note the Pu and Np from UNF was co-extracted and co-stripped.

- U, Pu and Np: 99.99% to the product streams/0.01% to the waste stream;
- Am and Cm: 99.9% to the product stream/0.1% to the waste stream;
- All remaining actinides and fission products: 100% to the waste stream.

The 1<sup>st</sup> and 3<sup>rd</sup> reprocessing facilities are assumed to handle used UO<sub>2</sub> fuel only. Both plants preferentially draw material from the current and historic nuclear used fuel buffers before drawing any UNF from the future UO<sub>2</sub> reactor fleets. The 2<sup>nd</sup> and 4<sup>th</sup> reprocessing facilities are assumed to handle used MOX fuel; any remaining capacity can then reprocess used UO<sub>2</sub> fuel. These units preferentially draw material in the following way:

- 1<sup>st</sup> priority: LWR MOX used fuel ponds
- 2<sup>nd</sup> priority: current and historic nuclear used fuel ponds
- 3<sup>rd</sup> priority: used fuel from future reactor fleets

The fissile material is then used as feed for Pu-U-Np metal fuel for the fast reactor fleet. Any additional material is available to fuel the LWR MOX fleets. At most, 33% of the LWR MOX cores fuel throughput is available for MOX fuel.

#### *Section 4 – 2nd generation (“new LWR”) new build LWR fleet*

A second generation of new build LWR plants is assumed to ramp up between 2030 and 2070, building up to a total capacity of 180 GWe. A 1300 MWe PWR plant design, based on a scaled down version of US-EPR was assumed here. In order to fully utilize the separated U-Pu-Np from the LWR reprocessing plants, the ORION model assumed any LWR in the fleet could be used for up to 33% MOX loading, if required to balance Pu-Np demand with production from the recycle plants. From 2040 to 2050, all these 2<sup>nd</sup> generation new build reactors would need to load 33% MOX in order to ensure the amounts of unused recycled plutonium remained low. Note however the fast reactor fleet was preferentially loaded with Pu-U-Np fuel with any remaining Pu-Np being used to fabricate MOX fuel for the LWR fleet.

For the ORION scenario there was a requirement to include americium & curium (Am-Cm) targets in some of the 2<sup>nd</sup> generation LWRs. A separate reactor node is included in Section 4 for this purpose. A separate reactor node was necessary because the presence of Am-Cm targets is expected to perturb the neutron spectrum and this affects the used fuel inventory model in ORION. Various assumptions were made in specifying the Am-Cm LWR node that were considered reasonable, but should be considered very speculative and uncertain given the current limited knowledge of Am-Cm targets in LWRs.

#### *Section 5 – Fuel Fabrication Plants*

Section 5 comprises a total of 13 fuel fabrication plant nodes that supply fuel to the various plant types; every distinct combination of reactor and fuel type has its own node. This improves traceability in the subsequent material flows in ORION and does not imply that 12 actual physical plants would be needed. All MOX fabrication plants assume a loss factor of 0.01%. Therefore 99.99% of all U, Pu and Np passing through the MOX fabrication plants leave as finished fuel and the remaining 0.01% is directed to waste. For the Am-Cm fabrication plant, the same loss factor of 0.01% is assumed for every nuclide.

#### *Section 6– fast reactor (ARR)*

A fleet of Advanced Recycle Reactor (ARR) plants is assumed to begin operating with a single demonstration plant in 2025, with full-scale deployment from 2045 building up to a final capacity of 97.6 GWe. These are nominally sodium cooled fast reactors fuelled with U-Pu-Np and are modeled in ORION making reasonable, but necessarily tentative, assumptions. A plutonium content of 18 w/o was assumed as the fresh fuel composition of all the ARR units, with adjustments using effective fissile coefficients to allow for the isotopic evolution of the plutonium with time during the scenario.

#### *Section 7 – Additional Reprocessed Waste Cooling*

All fission products modeled and non-recyclable material are cooled for an additional 70 years before being sent to a repository which in this model is called “Yucca Mountain” (middle right of Figure 2). The



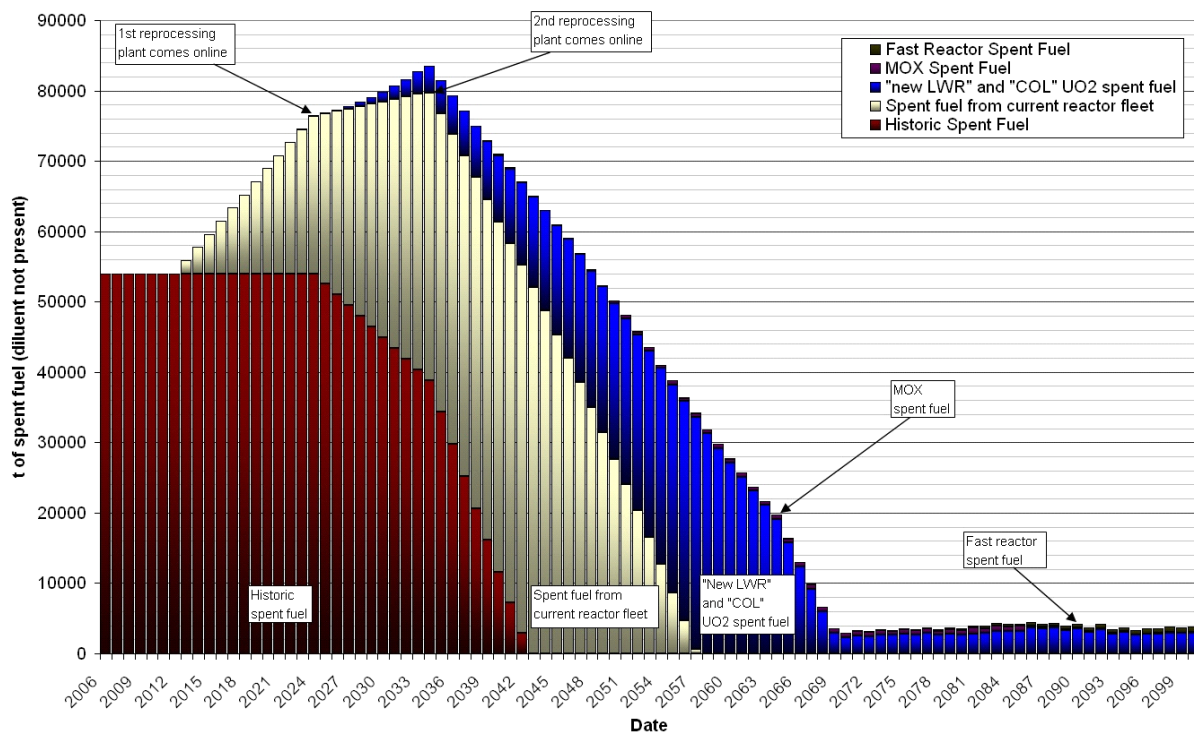
ES team’s analysis shows that other options for the final geologic repository become available when only the fission product waste is being disposed. Salt domes for example are eminently suitable and considerably less expensive to engineer.

*Section 8 – CANDU Operations*

The specification of the scenario made provision for the re-use of reprocessed uranium, and also for the irradiation of Am-Cm targets, in CANDU reactors. This is what the nodes in Section 8 represent.

**SAMPLE RESULTS FROM THE ORION SCENARIO MODEL**

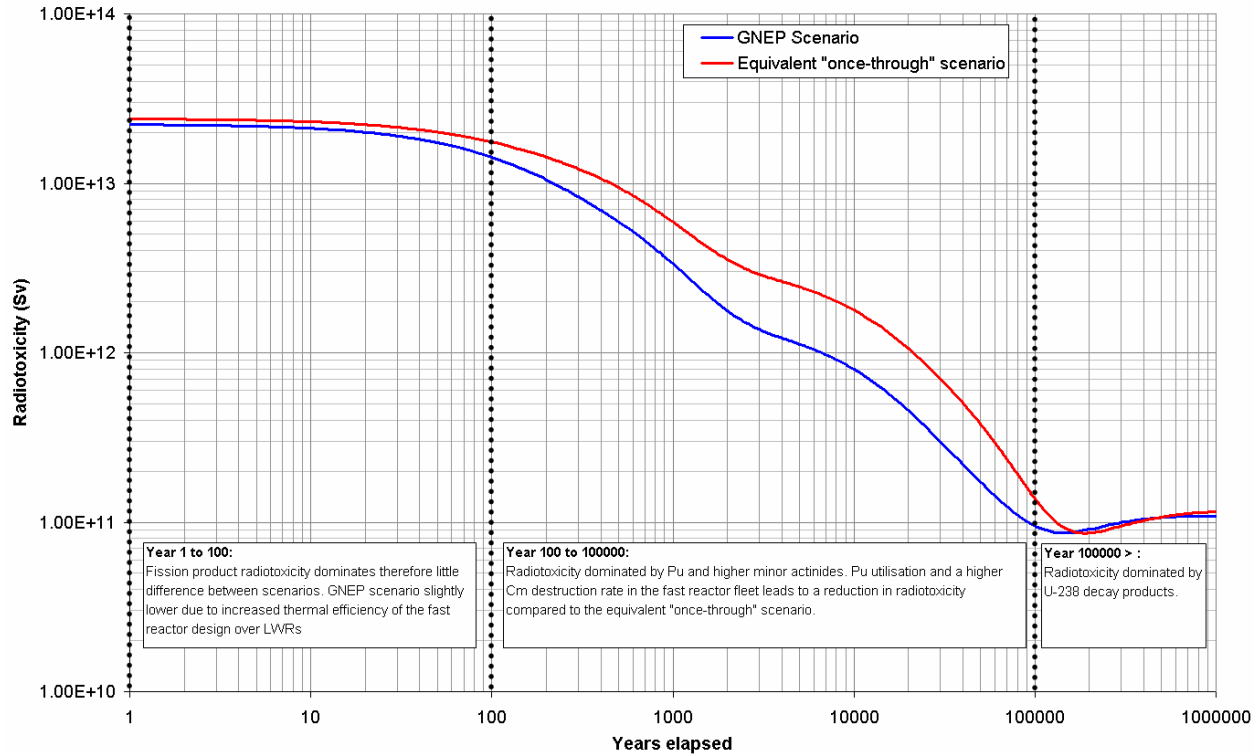
A small selection of results from ORION are shown here for illustration. Figure 3 shows the cumulative totals of used fuel masses from the different reactor fleets. As the various recycling plants come on-line, the cumulative mass of used fuel starts to fall and by 2044 all of the historic used fuel has been recycled. By 2070, the only used fuel is that from the ARR component which is cooling prior to recycle.



**Figure 3: Cumulative mass (tHM) of used fuel from the various reactor fleets**

The blue curve in Figure 4 shows the evolution of the total radiotoxicity (Sv) for the GNEP scenario. The red curve shows the radiotoxicity evolution for the equivalent “once-through” scenario in which the same nuclear output is derived from PWRs with UO<sub>2</sub> fuel without recycle. The GNEP scenario shows a significant reduction in radiotoxicity and a similar trend is also obtained for decay heat (not shown). Figure 5 shows the mean isotopic composition of the plutonium recovered from the recycle plant. At the start of recycle operations in 2025, the Pu-239 content is about 66%, but the mean falls after 2040 and stabilizes at about 45%. The Pu-238 content gradually rises from about 2% to about 7%, partly because of the neutron captures on the Np-237 co-product. This is important data for assessing the proliferation risk associated with the scenario, with the combination of the fissile Pu-239 decreasing and heat-producing Pu-238 increasing, being beneficial in this respect.





**Figure 4: Evolution of radiotoxicity content in the GNEP scenario**

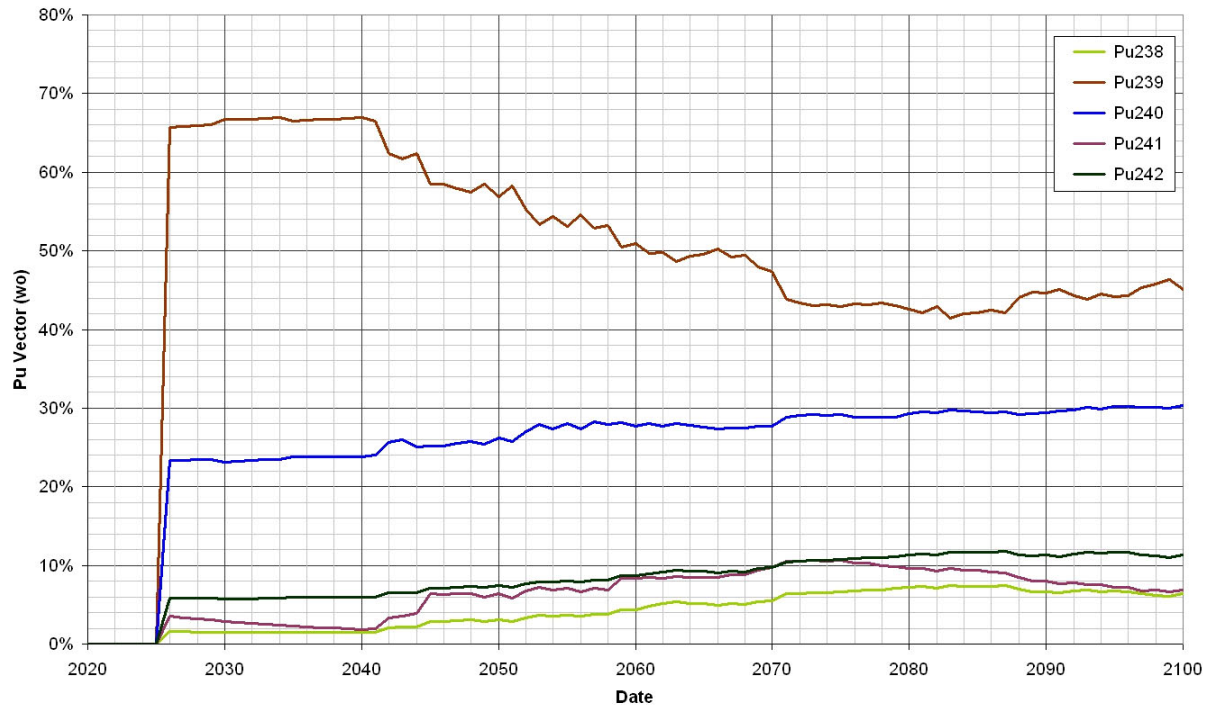
In addition to the mass flows for individual nuclides, other derived data, such as decay heat outputs and neutron emissions are recorded by ORION for feed fuels and waste products, but there is insufficient space to discuss them in this paper.

## PROLIFERATION RESISTANCE ASSESSMENT

Central to GNEP is the requirement for the recycle of fissile material for fuel sustainability. Conventional recycle schemes based on the PUREX process, as implemented in France, Japan and UK, involve the separation of chemically pure PuO<sub>2</sub>, which is considered undesirable in GNEP because of its potential mis-use for weapons proliferation. The GNEP fuel cycles currently being considered address this concern by separating mixed intermediate products, either U-Pu or U-Pu-Np, which will reduce the attractiveness for weapons proliferation. There is a clear need to evaluate quantitatively how effective such strategies are for improving proliferation resistance, but the difficulty is that there is no agreed international approach to this question.

NNL has previously developed a methodology for objectively and quantitatively assessing the relative proliferation resistance of different fuel cycles and this report describes its application to the GNEP fuel cycles. The method attempts to take account of not just the intrinsic proliferation resistance of nuclear material, but also the impact of the various protective barriers that are in place to prevent mis-use of nuclear materials. The development of the method was influenced by the IAEA Innovative Nuclear Reactors and Fuel Cycles (INPRO) Project [3] that recommended that proliferation resistance assessments for advanced reactor systems should be developed based on Scenario Analysis and/or Multiple Attribute Utility Analysis (MAUA) methods. However, INPRO stopped short of making specific recommendations and though there have been attempts by IAEA to develop proliferation metrics that could be agreed as an

international standard, so far it has proved difficult to establish a consensus. NNL’s approach provides a practical implementation of MAUA that can be applied to any nuclear fuel cycle.



**Figure 5: Evolution of the isotopic composition of recycled plutonium**

***Basic principles***

Multiple Attribute Utility Analysis (MAUA) is useful in engineering as a tool for selecting between different engineering options [4]. It involves the assignment of a utility function  $U(\underline{x})$  that is dependent on multiple attributes represented by the vector  $\underline{x}$ . The utility function would, for example, capture both the intrinsic utility value of an engineering component (eg cost, strength, mass, stiffness, corrosion resistance etc) and its availability. Thus if a potential new design for a component involved a material or a manufacturing process that offered technological advantages, but was still in need of extensive research and development, the R&D requirement would penalize the utility function under availability.

The proliferation risk of a reactor system lends itself to this approach. The utility function should reflect both the intrinsic value (proliferation attractiveness) of fissile or other radioactive materials and the probability of the material being accessed. There are many different pathways that might be considered and each of these would be assigned a separate utility function, but only one pathway, fissile material misuse by a state, is considered here. Specifically, this is taken to be clandestine access by a state to the fissile material that is active in the normal fuel cycle for the purpose of stockpiling nuclear weapons capable of producing a useful fissile yield.

For this pathway an intrinsic value can be assigned to the nuclear material for this purpose. It is proposed that a value function  $V(\underline{x})$  should be defined that captures the intrinsic value of the material for the specific illicit purpose. In this context, a high intrinsic value is regarded as undesirable and the value function should be as low as possible. Thus a reactor system  $s_1$  that produces only small quantities of poor

isotopic quality plutonium would be preferred over a system  $s_2$  which produced large quantities of high isotopic quality plutonium and the former's value function would be smaller (ie  $V_{s1}(\underline{x}) < V_{s2}(\underline{x})$ ).

The value function captures only the intrinsic value of the nuclear material relevant to the pathway and a separate access function  $A(\underline{x})$  quantifies how accessible the material is. For a given value function, a reactor system  $s_1$  with a large number of intrinsic and/or extrinsic barriers to accessing fissile material would be preferred to a system  $s_2$  with fewer barriers (ie  $A_{s1}(\underline{x}) < A_{s2}(\underline{x})$ ).

The utility function is the product of the value and access functions:

$$U(\underline{x}) = V(\underline{x})A(\underline{x})$$

A low value of the utility function can therefore be achieved by means of a small value function or a small accessibility function or (preferably) a combination of both. The method is insensitive to which means a particular system favors, so that equally acceptable proliferation resistance rankings could be achieved by a nuclear system which generates nuclear material with a high intrinsic value, but has large number of barriers, or by a system which has a smaller number of barriers but generates nuclear material of lower intrinsic value. Separating the utility function in this way is helpful because it simplifies the analysis. Although in principle both value and access functions could be functions of the complete attribute vector  $\underline{x}$ , in practice they are more likely to be functions of separate sub-sets of  $\underline{x}$  and this separability is thus a helpful feature of the method. The definition  $U(\underline{x}) = V(\underline{x})A(\underline{x})$  reflects the basic principle of security that the more intrinsically valuable a resource is, the more stringent are the security measures needed to protect it.

The approach is applicable to any pathway that could pose a threat of proliferation or diversion/theft. Although the full analysis considers several such pathways and each pathway has its own specific value and access functions, space only permits one to be illustrated here. This is the threat of state diversion or misuse of fissile materials for a clandestine weapons program:

In this case the value function should reflect the mass and isotopic quality of fissile material in the fuel cycle, to account for variations of critical mass, neutron output and heat output with isotopic characterisation. Various value functions have been reported in the literature [5] [6] for this purpose that involve various combinations of effective multiplication factor, critical mass, decay heat output and surface dose. The approach used in [4] is particularly simple and is adopted here, with the value function being defined by:

$$V(\underline{x}) = A \times (m_{Pu}/8) \times m_{fuel} \times \alpha^3 / [DH \times SF]$$

where  $\alpha$  is the so-called alpha-Rossi =  $(k-1)/\Lambda$ , where  $k$  is the  $k$ -effective at maximum compression and  $\Lambda$  is the prompt neutron lifetime in seconds. The normalization factor  $A$  is defined arbitrarily such that 8 kg of weapons grade plutonium (which constitutes a Significant Quantity (SQ), according to the definition used by IAEA) has a  $V_1$  of 1.0. The  $\alpha^3$  term approximates to the ratio {fissile yield}/{design yield} [7] for a Trinity-type device and therefore crudely captures the dependence of yield on the degree of super-criticality of the fissioning material.  $DH$  is the decay heat output in W/kg and  $SF$  is the spontaneous fission rate in neutrons/kg/s. These various factors in the value function have been estimated for the Pu-Np or U-Pu-Np co-product from the recycle plants as indicated by ORION.

The access function is more difficult to evaluate rigorously; it would be necessary to know the probability of being able to successfully breach every physical and institutional barrier that is in place. None of these is easy to evaluate in this context, because of the sensitive nature of the information. For example, although the probability of breaching each of the multiple security barriers designed to prevent terrorist

access to nuclear materials will have been assessed for each facility and may have widely different values (for example, turnstile access control, versus plutonium stored in a vault), such information cannot be expected to be released openly for this purpose. Similarly, while the effectiveness of physical barriers and IAEA safeguards to prevent clandestine access to nuclear materials by a state may have been assessed by IAEA, they again cannot reasonably be expected to be made available for use here.

Instead, a simplistic approach is proposed whereby each barrier, whether physical or institutional, is treated on an equal footing and the Access Function is assigned a value that depends only on the total number of barriers  $n$ .

$$A(n) = \epsilon^n$$

The barrier attenuation factor  $\epsilon$  is arbitrarily taken to be 0.1. This says nothing about the vulnerability of any security barrier or technological barrier, but is simply an arbitrarily assigned number that is used to discriminate between different systems. The important point is that for reasonable values of  $\epsilon$ , the relative rankings of different systems is unaffected by the precise choice. The only thing that matters is that  $A(n)$  does explicitly account for the intuitive fact that more barriers are better than fewer barriers and that each extra barrier reduces the utility function.

#### ***Application to GNEP advanced fuel cycles***

This section presents an analysis of the utility function for four advanced fuel cycles being considered for GNEP and the standard PWR recycle schemes used as reference points. There are therefore five cases in total:

1. ***PWR recycle reference case.*** This is the conventional PUREX reprocessing cycle as practiced currently in France, UK and Japan in which chemically pure separated  $\text{PuO}_2$  is produced as the intermediate which is re-used as Mixed Oxide (MOX) fuel.
2. ***PWR Pu-Np recycle.*** This is a case being considered under GNEP in which the pure separated plutonium is avoided by the producing a plutonium/neptunium mix in a ratio of 90:10 by weight.
3. ***PWR U-Pu-Np recycle.*** This is also one of the cases being considered under GNEP in which the Pu-Np mix is diluted in uranium, with an assumed 50:50 blend of U and Pu-Np by mass. The plutonium neptunium mix is again assumed to be in the 90:10 ratio, giving an overall mix for U-Pu-Np of 50:45:5 by weight. The 50:50 uranium/(Pu+Np) ratio assumed was chosen to be at the lower end of the range possible from NUEX, which can be up to 500:50 uranium/(Pu+Np).
4. ***Advanced Recycle Reactor (ARR) Pu-Np recycle.*** Same as 2 but applied to ARR fuel cycle.
5. ***ARR U-Pu-Np recycle.*** Same as 3 but applied to the ARR fuel cycle.

The GNEP fuel cycles should rank more favorably than the conventional PWR recycle reference case, because of the specific features introduced to improve the inherent proliferation resistance.

The various value functions were calculated using the actual isotopic makeup of the intermediate streams using the decay constants, decay heat outputs, spontaneous neutron emissions and inhalation effective dose coefficients tabulated for the individual nuclides. The value function also requires knowledge of the alpha-Rossi parameter  $(k-1)/\Lambda$  for an implosion device at maximum compression. This requires knowledge of the infinite multiplication factor of the metal and of the neutron leakage, both at the point of maximum compression, neither of which are openly available because of the obvious sensitivities. Calculations of the infinite multiplication factors were made using a lattice code. Then estimates were made of the neutron leakages at maximum compression using what are essentially little more than dimensional arguments. Although the alpha-Rossi estimates so obtained are very approximate, there is actually relatively little sensitivity of the value function to this parameter and there is certainly no impact on the relative rankings of the value functions.

Because the NUEX recycle plants in the GNEP scenario produce a Pu-Np co-product, there is a significant proliferation resistance benefit because the consequential Pu-238 production reduces the value function. However, although not considered in this paper, it should be noted that the full assessment also includes other risk pathways such as diversion/theft by a sub-national organization for the purpose of radiological dispersion. Since Pu-238 is the main contributor to inhalation toxic potential, an elevated Pu-238 inventory can have a deleterious consequence for other such pathways, emphasizing the complex considerations that need to be taken into account in the overall analysis.

### 5.2 Access functions

For each of the five systems considered, Table 2 identifies the number of access barriers for clandestine access by a state to fissile material, which determine the access functions. In the conventional PWR recycle case, the only barrier to state access is considered to be the IAEA safeguards. For the other cases considered as part of the GNEP scenario, the presence of Np-237 and its associated radiation field counts in this analysis as an additional barrier. If for a clandestine weapons program the Np-237 was not separated out from the co-product, it would complicate the weapon design and assembly. Alternatively, if it was deemed necessary to separate the Np-237, this in itself constitutes an additional complication that would require the construction of a facility for this purpose. Either way, the presence of the Np-237 in the co-product can be taken as constituting a barrier in this sense. A similar argument would apply to the presence of the uranium in the U-Pu-Np co-product case. Not separating the uranium from the co-product complicates weapon design (larger critical mass), while separating the uranium requires a plant for that purpose.

**Table 2 : Access function parameters for clandestine access to fissile material by a state**

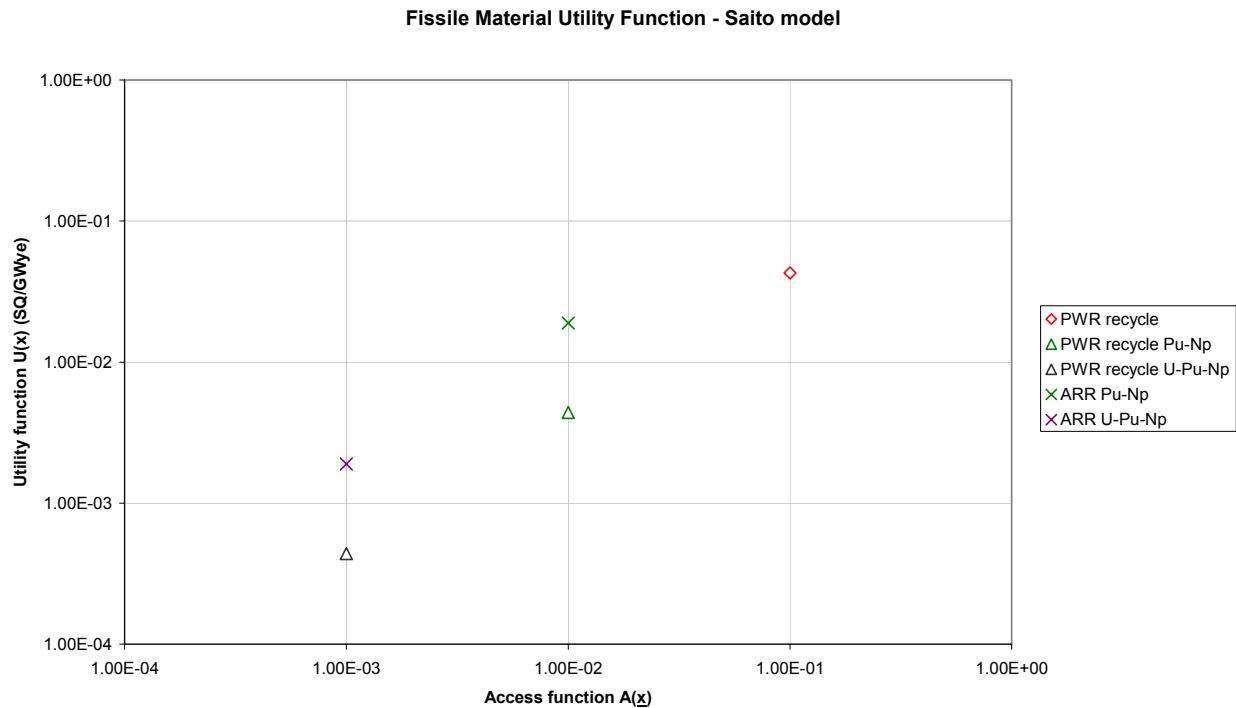
| System              | Number of barriers n | Barriers   |
|---------------------|----------------------|--|
| PWR recycle         | 1                    | IAEA safeguards  |
| PWR recycle Pu-Np   | 2                    | IAEA safeguards<br>Np-237 alpha radiation                  |
| PWR recycle U-Pu-Np | 3                    | IAEA safeguards<br>Np-237 alpha radiation<br>Co- product U |
| ARR Pu-Np           | 2                    | IAEA safeguards<br>Np-237 alpha radiation                  |
| ARR U-Pu-Np         | 3                    | IAEA safeguards<br>Np-237 alpha radiation<br>Co-product U  |

Figure 6 shows a plot of the utility function on the y-axis versus the access function for misuse/ clandestine access by a state to fissile material. The units of the utility function are SQ/GWye.

Plotting this function against the access function is helpful because it shows immediately the effect of the access function. All four cases with Pu-Np or U-Pu-Np co-product show an improvement relative to the conventional recycle case, largely because of the presence of the additional barriers that reduces the access function. The two ARR points have higher utility functions than the PWR cases because ARR has a higher mass of plutonium per GWye in the active fuel cycle than the PWR fuel cycle because of the high initial loading of plutonium in the ARR fuel assemblies. Figure 6 shows the clear proliferation resistance benefit of using the Pu-Np or the U-Pu-Np co-products.

Once the Pu-Np or U-Pu-Np co-product is irradiated having been recycled, the presence of elevated levels of Pu-238 in the scenario is generally beneficial, decreasing the value and utility functions of the fissile material recovered in any subsequent recycle. Though the alpha-Rossi parameter is slightly lowered by the high Pu-238, the main impact is through its high contribution to decay heat.

It is instructive to compare these results with those from the study [6] on proliferation attractiveness carried out by Los Alamos, Lawrence Livermore and Pacific Northwest National Laboratories in the US. In this study, the presence of uranium at a 2.3/1 U/Pu ratio is assessed as providing a proliferation attractiveness level of “C”, where attractiveness level “A” is a nuclear weapon and “E” is highly active waste or uranium containing less than 20% U-235. To obtain a proliferation attractiveness of “D” this study concludes that a 5.2/1 U/Pu ratio is required. The study also concludes that the presence of neptunium is not helpful in reducing proliferation attractiveness. The difference between the conclusions of this US study and the one presented in this paper can be explained by recognizing that the US study concentrated on proliferation attractiveness, which is equivalent to the Value Function in this paper. It does not take into account what is referred to in this paper as the Access Function, which assesses the number of barriers to clandestine access to fissile material.



**Figure 6: Utility function versus access function plot for fissile material mis-use by a state**

## CONCLUSIONS

EnergySolutions and its team partners have carried out detailed design studies for US Department of Energy in support of GNEP. Central to the analysis is a detailed scenario model for a future US fuel cycle based on used nuclear fuel recycle. The NUEX separations chemistry scheme developed by EnergySolutions produces a Pu-Np or U-Pu-Np co-product that avoids the separation of pure PuO<sub>2</sub>. The NUEX process is a modification of the well known PUREX process that can be implemented on a commercial scale with the minimum of development work and that will pose minimal technological risk.

The analysis has produced detailed mass flows for the scenario which were determined using NNL's ORION scenario analysis program. ORION is an invaluable tool for this purpose, because it allows the time evolution of a realistic (and necessarily complex) reactor scenario to be modeled. Previously, it has often been the case that future reactor scenarios have been modeled on the basis of idealized scenarios in which different reactor fleets are assumed to be operating in an equilibrium condition. While it can yield some valuable insights, the difficulty with this equilibrium approach is that it is often not practically attainable in the finite timescales of realistic scenarios. This was in part the motivation for the ES Team and NNL to model the US reactor fleet in detail, as it provides a sound context in which to evaluate the performance of the favored U-Pu-Np recycle scheme. Some sample results from ORION have been presented to illustrate the kind of outputs that it able to provide.

The final part of the paper has presented an attempt to assess the effectiveness of using the Pu-Np or U-Pu-Np co-product to improve the inherent proliferation resistance of the recycle scheme, compared with the conventional recycle scheme that produces separated PuO<sub>2</sub>. A method is described which attempts to quantify the proliferation risk in an objective manner, taking account of both the intrinsic value of the fissile material in a proliferation scenario (guided by the outputs from ORION) and the accessibility of the fissile material in the scenario. Compared with conventional PUREX recycle, the method demonstrates a clear advantage of using the Pu-Np and especially U-Pu-Np co-product.

Overall, this fuel cycle modeling work has demonstrated that the GNEP goals can be achieved, that recycling of UNF, re-use of recycled uranium, re-use of MOX and burning of americium and curium in thermal reactors can provide a means of dealing with UNF that is realizable in the short term. The model provides a sound basis for assessing the economics of this program.

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