

ASSESSMENT OF PROLIFERATION RESISTANCE REQUIREMENTS FOR FAST-REACTOR FUEL-CYCLE FACILITIES

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ABSTRACT

Inclusion of intrinsic safeguards in fast reactor systems could lower one of the barriers to a closed fuel cycle. This project aims to locate and evaluate the proliferation concerns in a generic fast reactor fuel cycle: plutonium driver fuel from LWR or CANDU spent fuel with a depleted uranium blanket and PUREX reprocessing. Proliferation Resistance (PR) measurement methodologies have rarely, and never quantitatively, been applied to fast reactor systems. A literature review has been completed of the known fast reactor PR evaluations and additional PR methodologies applied to other reactor systems. Quantitative estimates for the material flow in a fast cycle were developed. The GEN IV International Forum suggested Multi Attribute Utility Analysis (MAUA) methodology for its semi-quantitative approach. From these reviews, estimates, and suggestions, this project applies a multiplicative MAUA methodology of assessing PR to establish the points in the fuel cycle of most interest. These areas of interest are analyzed in more detail and a methodology for the inclusion of intrinsic safeguards in these areas is being developed. Threat-scenario selection and metrics are along GEN IV PR&PP methodology guidelines and allow for quantitative trade-off studies as envisaged by GEN IV International Forum.

Keywords: Fast Breeder Reactor, Closed Fuel Cycle, Proliferation Resistance (PR), Multi Attribute Utility Analysis (MAUA).

INTRODUCTION

Fast breeder reactors with a closed fuel cycle (FBRFC) are important to the sustainability, reliability, and security of the world's long-term energy supply. The attractiveness of the FBRFC includes the conversion of the abundant fertile isotope ^{238}U into fissile material ^{239}Pu at rates faster than it is consumed (breeding). Fast reactors have a hundred-fold energy extraction potential from the same amount of mined uranium compared to thermal reactors and have the possibility of incinerating all long-lived heavy elements during reactor cycle. Knowing the vast potential of the FBRFC technology, countries such as Japan, India, and China are actively pursuing research in this area to develop advanced commercial fast reactor energy systems^{1,2,3}. The United States, France and Japan have recently announced to increase their cooperation on sodium-cooled fast reactor prototypes⁴. Also, the GEN IV International Forum (GIF) proposal has three nuclear energy systems based on FBRFC; the gas cooled fast reactor system (GFR), the lead cooled fast reactor system (LFR) and the sodium cooled fast reactor system (SFR)⁵. However, the breeding of high purity ^{239}Pu isotope and its envisaged use in large quantities in FBRFC by design is a major safeguards concern, because of the vulnerability to possible special nuclear material (SNM) diversion from peaceful uses to destructive ones. Hence, it is prudent to assess the proliferation resistance (PR) of the FBRFC when addressing other issues such as radioactive waste generation, economics and safety.

The Nuclear Security Science and Policy Institute (NSSPI) at Texas A&M University has now taken up a research project to conduct pre-conceptual design studies for the integration of modern safeguards directly into the planning and building of FBRFC facilities. The project is taken up under the sponsorship of DOE National Nuclear Security Administration's office of Non-proliferation and International Security. The project tasks are to (a) identify a suitable PR assessment methodology from the available methodologies through literature survey, (b) develop nuclear material flow diagram of a generic FBRFC by computational efforts and (c) perform the PR assessment for different diversion or misuse scenarios. The results of the project study should lead to high-priority tradeoff studies, identify the possible weak links in the FBRFC, and suggest specific ways to strengthen them by the integration of modern safeguards. Finally, the expected results should aid the International Atomic Energy Agency (IAEA) or domestic inspecting entity to effectively and efficiently monitor and verify SNM in a manner that provides minimal intrusion into normal facility operations. The paper here discusses (1) excerpts from the literature review carried out on FBRFC and PR assessment methodologies and (2) computational efforts for the FBRFC material flow diagram development.

LITERATURE REVIEW

The FBRFC^{6,7,8}

The major facilities in a generic FBRFC are shown in Figure 1: (1) the fuel fabrication unit, (2) the Fast Breeder Reactor (FBR), and (3) the spent fuel reprocessing facility, which includes an interim spent fuel storage and a waste management facility on site. The FBR may be fuelled with either highly enriched uranium or plutonium. Because fission cross-sections are much lower in the fast neutron energy domain than in the thermal region, fuel must contain higher concentrations of fissile material. The proportions of fissile uranium or plutonium in fast-reactor

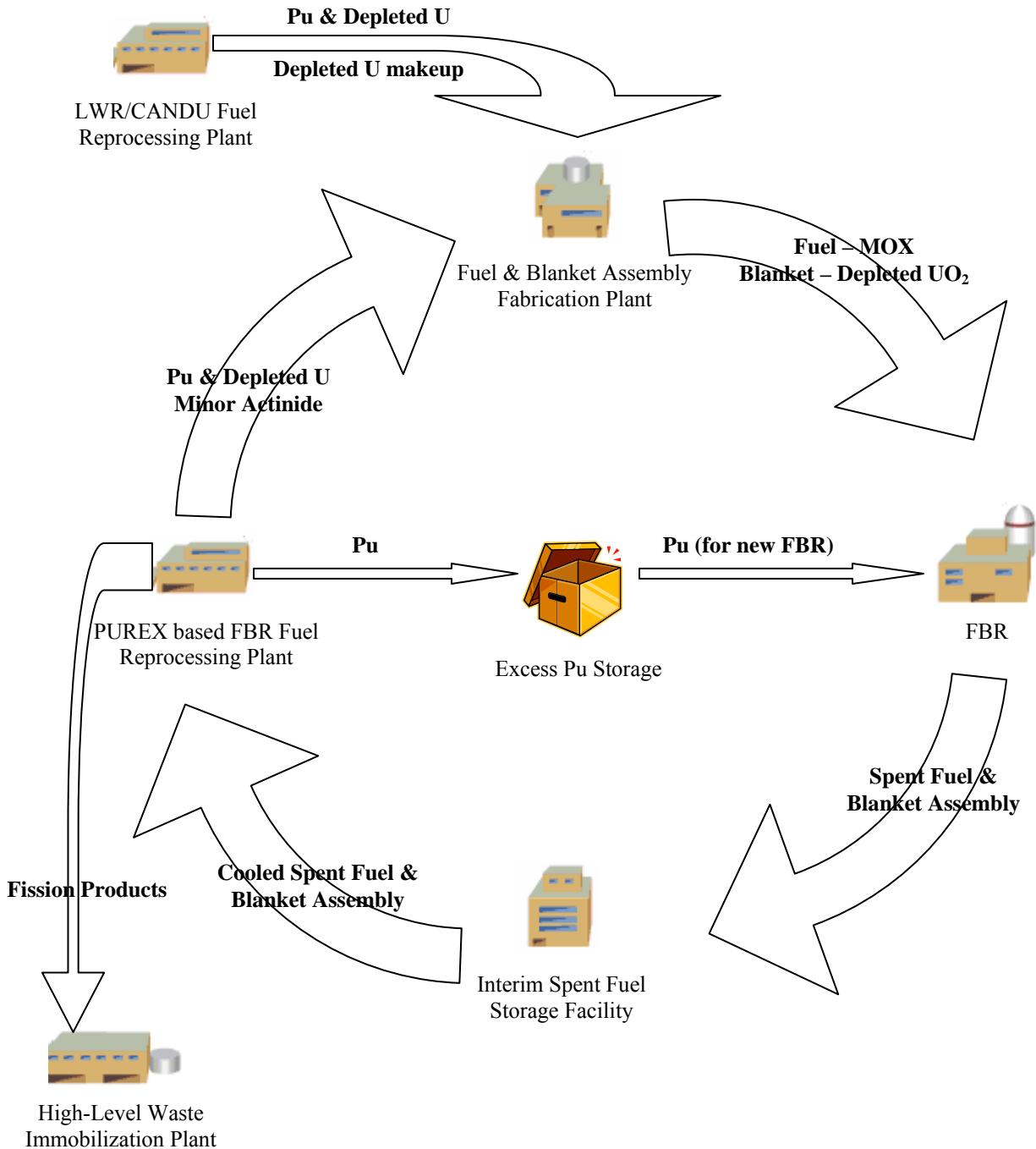
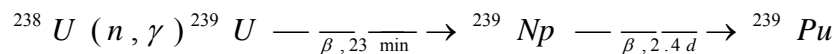


Figure 1: Schematic of Fast Breeder Reactor Fuel Cycle

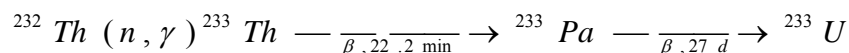
fuel therefore range from 15% to 30% as compared with 3% to 5% in thermal reactors. The FBR cores are also less uniform than thermal-reactor cores. FBR cores are compact and are typically divided into three major regions, namely the inner core region, outer core region and the blanket region. Fuel cores are loaded with driver assemblies in the inner and outer core regions and are typically surrounded by arrays of depleted or fresh uranium fuel assemblies (blankets) in which plutonium is produced by neutron capture. The average core neutron energy spectrum will be in

the range of a few hundreds of keV compared to that of a few eV in thermal reactors. An FBR operates with high power density and uses a liquid metal coolant. The fuel burn-up achieved is also very high in FBR, often three times the normal maximum burn-up of thermal reactors. The current project aims at studying FBR cores fueled with plutonium driver assemblies in which plutonium (15% to 30%) is mixed with uranium (85% to 70%) for the reasons aforementioned. Oxide, carbide, nitride or metallic fuel can be used, in increasing order of usefulness in plutonium breeding. Confining the study to Mixed Oxide (MOX) fueled cores, the plutonium enrichments are ~20% and ~30% in the inner and outer core regions respectively. The varied enrichments are to maintain nearly a uniform radial power distribution. Usually depleted uranium is loaded in the axial and radial blanket regions. Plutonium is produced in both blanket and core region through the following reaction.



FBR core designs are made in such a way that the formation of plutonium in the core as well as in the blanket region will exceed the amount of plutonium being consumed for power production. The important point to note here is that the plutonium bred in the core region will be reactor grade, but blanket regions will breed weapon grade material. This is because the blanket region is less exposed to neutron flux, which in turn substantially reduces the conversion of ${}^{239}\text{Pu}$ isotope into higher mass isotopes such as ${}^{240}\text{Pu}$, ${}^{241}\text{Pu}$ and ${}^{242}\text{Pu}$.

The only existing commercially viable option to recover the unspent plutonium and uranium is by reprocessing the spent fuel and blanket assemblies using the solvent extraction process known as PUREX (Plutonium URanium EXtraction)^{9,10}. The PUREX process used in thermal reactor spent fuel reprocessing requires minor modifications to be incorporated so that it can be applied for fast reactor spent fuel reprocessing. Modifications are essentially needed to take into account the greater plutonium inventories and radioactivity content produced at higher levels of burn-up. Only in France and the UK has fast-reactor fuel been consistently reprocessed and separated plutonium recycled. Although a fast-reactor fuel reprocessing facility is currently available in Japan, the non-operation of their fast reactors JOYO and MONJU has halted Japanese plutonium recycling for fast reactors. India has demonstrated the reprocessing capability of fast reactor fuels in its pilot plant in the year 2003. India successfully reprocessed the plutonium-uranium carbide spent fuel discharged from its fast breeder test reactor (FBTR), which had a burn-up of 100 GWD/T. India has immediate plans to build bigger reprocessing facilities to reprocess spent fuel from its proto-type fast breeder reactor (PFBR-500MWe) to be commissioned by the year 2010. Another interesting aspect is that thorium can be employed instead of depleted uranium in the blanket regions. In that case the fissile material produced in the blanket regions will be ${}^{233}\text{U}$ instead of ${}^{239}\text{Pu}$ according to the following reaction.



The ${}^{233}\text{U}$ produced is always contaminated with ${}^{232}\text{U}$, which in turn decays to hard gamma emitting daughter products like ${}^{212}\text{Bi}$ ($\gamma = 0.7$ to 1.6MeV) and ${}^{208}\text{Tl}$ ($\gamma = 2.6\text{MeV}$). Even though this material is self-protecting from the non-proliferation point of view, the levels of ${}^{232}\text{U}$

contamination needs to be examined because it can largely vary with blanket-irradiation times and the neutron spectra encountered in the blanket regions.

PR Assessment Methodologies

A Literature survey was performed to explore the available PR assessment methodologies proposed by different institutions. Three independent reviews by Krakowski¹¹, Takaki *et al*¹², and Giannangeli¹³ high-light the available methodologies put forth by IAEA (**INPRO** - International PROject on innovative nuclear reactors and fuel cycle)¹⁴, GEN IV experts' group (**PRPP** - Proliferation Resistance and Physical Protection)¹⁵, **AFCI** (Advanced Fuel Cycle Initiatives) multi-attribute utility analysis (**MAUA**) methodology¹⁶, JAEA's **FS Project** (Feasibility Studies on commercialized fast reactor cycle system)¹⁷, **TOPS** (Technological Opportunities to increase the Proliferation resistance of global civilian nuclear power Systems methodology)¹⁸, **BNL** (Brookhaven National Lab methodology)¹⁹, **SNL RIPA** (Sandia National Lab Risk Informed Probabilistic Analysis)²⁰, and **SAPRA** (Simplified Approach for PR Assessment of nuclear systems)²¹. In addition to the above methods, special attention has been paid to PR analyses on FBRFC systems. There exist four relevant cases. The analysis by Ahmed *et al*²² used the multi-attribute decision model to find the potential routes of nuclear proliferation. Eleven routes were considered in their study including the FBR route. Heising *et al*²³ also used the multi-attribute decision model with three different nuclear energy systems, vis-à-vis once through LWR, LWR coupled with FBR and a thorium cycle coupled with an advanced converter breeder system. Both of the above studies were concluded about 25 years ago. A recent PR assessment work carried out for the Russian fast reactors BN-600 and BN-800 applying the MAUA based methodology for tightly coupled plutonium recycling between fast reactors and reprocessing facility is found in Zrodnikov *et al*²⁴. Finally, the ongoing Japanese initiative to address issues in assessing and maximizing PR for future commercial FBR systems (to be completed by 2015) is reported by Sagayama²⁵.

It is seen from a literature survey that the use of MAUA methodology for PR assessment is wide spread. The MAUA approach is a well-established decision analysis methodology and has evolved since the first publication on the topic in 1978. MAUA consists of several methods of compiling multiple factors in order to make a single decision and has the ability to incorporate complex and inter-related components in a decision. Each factor has impacts on risks, resources, timelines, and/or levels of effort associated with the acquisition of a significant quantity (SQ) of SNM. This can be accommodated in a MAUA analysis to assess relative PR among various viable options.

It is also worthwhile to mention here that the application of the MAUA methodology for the PR assessment of FBRFC fits well into the PR&PP framework proposed by the GIF. The GIF PR&PP has identified threat scenarios for the PR assessment and has proposed six preliminary measures each for estimating the PR and PP of nuclear energy systems (NES). The proposed measures for PR are (1) Proliferation Technical Difficulty, (2) Proliferation Resources, (3) Proliferation Time, (4) Fissile Material Quality, (5) Detection Time, and (6) Detection Resources. The measures of PP are not elaborated here, since the current NSSPI project as a first step aims only at the PR assessment. Suggestions for selecting metrics for each of the above six

measures for carrying out the PR assessment is available in the report prepared by the GIF. For assessing the PR measures of a NES, the GIF proposes a three step progressive approach vis-à-vis, (a) qualitative assessment [coarse path level approach], (b) semi-quantitative assessment [MAUA type approach] and (c) full scale quantitative assessment. The final quantitative assessment needs extremely detailed flow paths of the NES which is not currently available for the FBR system under study. Hence, for the present project study a semi-quantitative assessment based on the MAUA methodology is more appropriate. However, the selection of threat scenarios and the metrics required for MAUA is made as per the guidelines of GIF methodology so that a pure quantitative assessment can be made in the future. This MAUA methodology will utilize the salient feature derived from the proposed SAPRA methodology to consider four separate proliferation stages: (1) diversion from a declared facility, (2) transportation to a covert facility, (3) material transformation to weapons-usable form, and (4) weapon fabrication.

The MAUA Methodology²⁶

The MAUA is a well-established decision analysis methodology and has evolved since its first publication in 1978. The MAUA consists of several methods of compiling multiple factors in order to make a single decision. The important task in MAUA methodology is to obtain an objective function, commonly referred to as the utility function, which incorporates multiple attributes that best describe the attractiveness of a system. For the system under study, material flow routes in a generic FBR fuel cycle are to be analyzed in terms of a PR value that lies between 0 (worst state) and 1 (best state). By scrutinizing the parameters associated with the fuel cycle, the analyst can find a utility function that can serve as a guide in decision making. The best decision, obviously, is that which maximizes the function's value.

Attribute values aid in ascertaining conclusions regarding the possibility of material diversion at a given rate under certain conditions of risk. These attributes are used in composite or weighted form to make a final assessment. Differences between MAUA analyses include the use of different weighting schemes, the use of various utility functions, and the selection of different attributes for inclusion. See Keeney and Raiffa²⁶ for details of the basic MAUA related results asserted below.

The various steps involved in the MAUA assessment are:

- Define utility function $u(x_1, x_2, \dots, x_i)$ to represent value of a given path for a range of attribute values x_i ;
- Define the single-attribute utility functions $u_i(x_i)$, that contribute to this overall utility;
- Define a set of attributes, $\{x_i\}$, that can be related to cost, time, material quality, or other characteristics deemed of value or utility;
- Assume preferential and utility independence of the chosen attributes, so that either of the following relationships is valid:

$$u(x_1, x_2, \dots, x_i) = \sum_1^n k_i u_i(x_i) \text{ (Pure additive form) } \underline{\hspace{10em}} \quad (1)$$

$$1 + Ku(x_1, x_2, \dots, x_i) = \prod_i^n (1 + Kk_i u_i(x_i)) \text{ (Pure multiplicative form) } \underline{\hspace{10em}} \quad (2)$$

The multiplicative MAUA method is more suitable for the current study due to the fact that the additive MAUA does not behave appropriately in limiting cases¹³. In equations (1) and (2), u and $u_i(x_i)$ are utility functions scaled from 0, for the worst state, to 1, for the best state; the k_i values are appropriately selected weights for the given attributes with $0 < k_i \leq 1$, and $K > -1$.

In Equation (3), $K = 0$, if $\sum_1^n k_i = 1$; otherwise K must be computed from the specific normalization conditions given in Equation (3).

$$1 + K = \prod_i^n (1 + Kk_i) \quad (3)$$

Since we wish to use the multiplicative utility function (Equation 2), the weighting factors must meet two requirements. First, the sum of all the weighting factors must not be exactly 1.0, because that would reduce the equation to the additive utility function. So the sum of all weighting factors will be selected as greater than 1. Also, the weighting factors cannot each be equal to 1.0. If they were, then the only solutions to Equation 2 would be $K = -1$ and $K = 0$, neither of which are valid. The solution must lie on $-1 < K < 0$ when $\sum_1^n k_i > 1$ (the multi-attribute utility function exhibits risk aversion). If, $\sum_1^n k_i < 1$, the multi-attribute utility function exhibits risk seeking, and hence will not be used. The solution to find K in the range $-1 < K < 0$ for the normalizing condition in Equation (3) for a typical case is demonstrated in Figure 2. In this case, 10 attributes x_i of equal weights k_i ($i=1, 10$) with $\sum_1^n k_i = 2$ are assumed for the utility function representing a particular path in the system. In the present PR analysis the route to nuclear proliferation as mentioned earlier is split into four different stages vis-à-vis (1) material

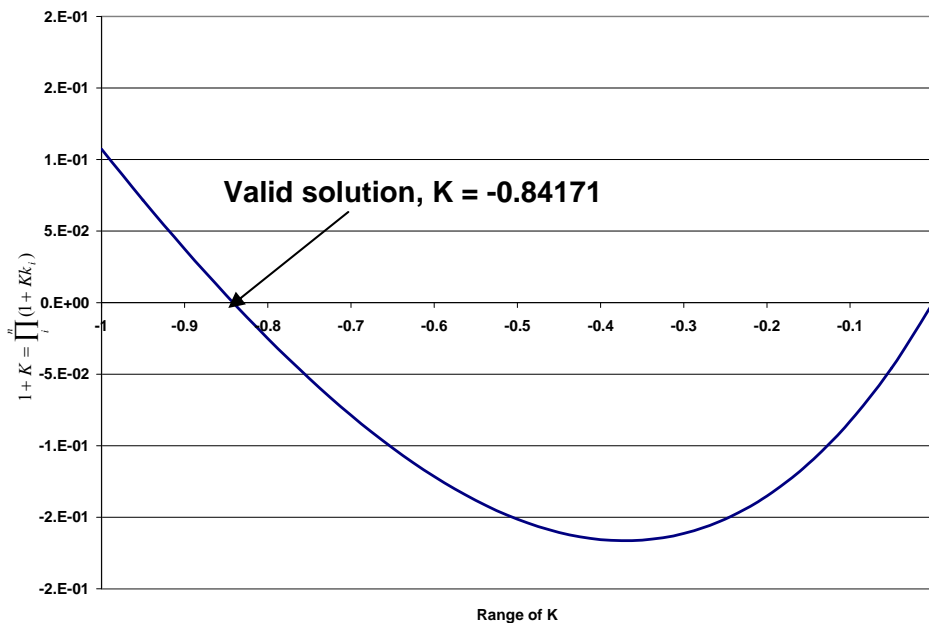


Figure 2: Solution of K from the normalizing condition

diversion, (2) material transportation, (3) material transformation and (4) weapons fabrication. The list of attributes for each of these stages of diversion for use in this study is taken from a recent report issued by Sandia National Laboratories (SNL) ²⁷. They are listed in Table 1. Some of the attributes listed here are specifically country/facility, material, or weapons-design based and not known at this point of time, as the study deals with a generic FBRFC. Some of the attributes are evaluated through computational work for a specific FBR core design, the details of which are elaborated in the following section.

Diversion Stage	32. Heat load of material
1. Mass/SQ of nuclear material	33. Shield thickness to reduce radiation to 10 mR/hr
2. Volume/SQ of nuclear material	34. Host country size
3. Number of items/SQ	35. Number of declared nuclear facilities
4. Material Form	36. IAEA imagery analysis rate
5. Radiation level in terms of dose	Transformation Stage
6. Chemical reactivity	37. Number of process steps to metallic form
7. Temperature of Source Process	38. Number of export controlled/equipment/materials
8. Heat load of material	39. Minimum electrical requirement
9. Uncertainty in accountancy measurements	40. Number of unskilled workers required
10. Expected vs. Actual MUF	41. Number of skilled workers required
11. Frequency of measurement	42. Number of advanced degree work
12. Amount of Material Available	43. Number of Technical Experts
13. Probability of detection	44. Additional Protocol in force?
14. Is there enough physical space to modify?	45. Environmental sampling rate
15. Number of People for Modifications	46. Sensitivity of IAEA equipment
16. Remote handling tools required?	47. Isotopic signatures
17. Specialized tools required?	48. Facility size
18. Requirement for the process to be halted for modifications	49. Heat load of transformation process
19. Risk of Modification (safety)	50. Sonic load
20. Risk of penetrating containment	51. Radiation load
21. Probability of being caught	52. Volume of non-naturally occurring gases emitted
Transportation Stage	53. Undiluted volume liquid emissions
22. Mass/SQ of nuclear material	Weapons Fabrication Stage
23. Volume/SQ of nuclear material	54. Spontaneous fission production Rate
24. Material Form	55. Radiation exposure at one meter
25. Radiation level in terms of dose	56. Heating rate of weapons material
26. Heat load of material	57. Can use ballistic assembly methods?
27. Chemical reactivity	58. Number of phases in the phase diagram
28. Immediate Chemical toxicity	59. Radiation level in terms of dose
29. Time Average Chemical toxicity	60. Chemical reactivity
30. Mass of material and transportation container	61. Radio-toxicity
31. Volume of material and transportation container	62. Knowledge and skill level for material/weapon type alternatives

Table 1: List of attributes selected for PR assessment

MATERIAL FLOW DIAGRAM DEVELOPMENT FOR FBRFC

A quantitative nuclear material flow through a generic FBRFC is required to estimate the values of the attributes for the PR assessment. In order to demonstrate the validity of the computational framework chosen for the study, a specific FBR core design is selected to develop the material

flow diagram. This will be made more generic in the future studies. The basic computational framework is to couple the well-validated neutronics code MCNP⁵²⁸ for 3D modeling of FBR core and the general point depletion code ORIGEN²⁹ for burn-up calculations of fuel and blanket assemblies, by using a PERL script developed under the name MONTEBURNS³⁰. A schematic flow diagram of how MONTEBURNS interacts with MCNP and ORIGEN is shown in Figure 3. Snap shots of FBR core model generated using the plot feature of MCNP code are shown in Figure 4. Following attributes are evaluated from these computations: (1) plutonium isotopic vector in fresh as well as irradiated fuel and blanket assemblies, (2) amount of plutonium and uranium to be fed into the equilibrium core of the FBR per year, (3) irradiated plutonium and uranium from FBR equilibrium core available for reprocessing per year, (4) gamma and neutron radiation source term and heating rate of the irradiated fuel and blanket assemblies, (5) number, weight and volume of the items to be diverted for obtaining a SQ.

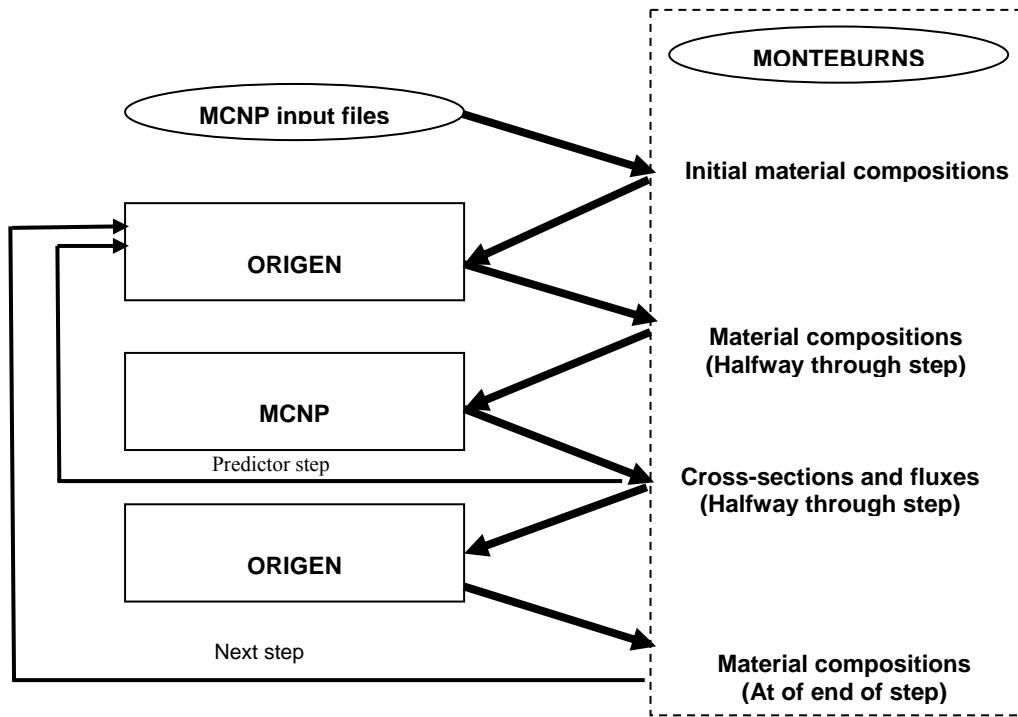


Figure 3: Interaction of MONTEBURNS with MCNP and ORIGEN

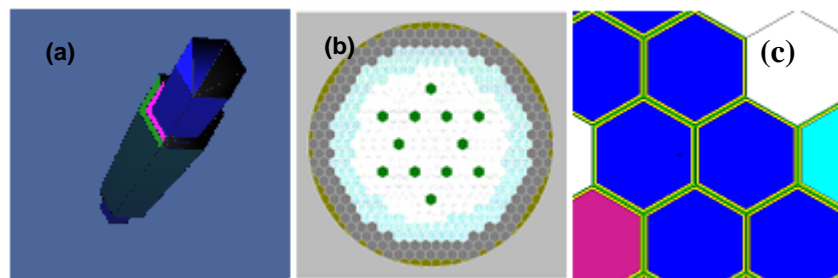


Figure 4: Plots obtained using MCNP modeling of a typical FBR core: (a) Fuel assembly in 3D (b) Full core at mid plane (c) Inner, outer core fuel and blanket assemblies

SUMMARY AND STATUS OF THE PROJECT

- A research project to conduct pre-conceptual design studies for the integration of modern safeguards directly into the planning and building of FBRFC has been initiated at NSSPI. Project study is divided into four tasks: (i) literature review, (ii) development of material flow diagram for a generic FBR fuel cycle, (iii) definition of relevant design space and prioritization of associated tradeoff studies, and (iv) preliminary tradeoff results for high-priority tradeoff studies.
- Literature survey has been completed on: (i) nuclear material flow diagram for a FBRFC and (ii) PR assessment methodologies. The underlying computational frame work uses state-of-art-computer codes MCNP5 to model FBR core and ORIGEN2.2 for burn-up calculations, coupled by MONTEBURNS to develop a quantitative material flow diagram for the FBRFC. The assumptions made for the FBRFC used for the present study are: (1) feed plutonium for FBR comes either from spent fuel of CANDU type reactors or LWRs; (2) PUREX is used for reprocessing FBR spent fuel; and (3) depleted uranium is employed in the blanket regions. This computational framework is needed to obtain the attribute values such as (a) plutonium isotopic vector in fresh as well as irradiated fuel and blanket assemblies, (b) the amount of plutonium and uranium needed for the equilibrium core of the FBR per year, (c) irradiated plutonium and uranium from FBR equilibrium core available for reprocessing per year, (d) gamma and neutron radiation source term and heating rate of the irradiated fuel and blanket assemblies, (e) number, weight and volume of the items to be diverted for obtaining a SQ of SNM, required for PR assessment.
- The semi-quantitative multiplicative MAUA approach will be used for PR assessment of FBRFC. However, the selection of metrics required for MAUA is made in line with GEN IV PR&PP methodology, so that a pure quantitative assessment is feasible for future studies.

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