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MODELING REPORT of DYMOND Code (DUPIC version)

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제출문

한국원자력연구소장 귀하

본 보고서를 건식재가공 핵연료 노심특성 기술개발 과제 수행중 수행된 기술보고서로 제출합니다.

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주저자 : 박주환 (건식공정핵연료기술 개발부)

공저자 : Abdellatif M. Yacout (Argonne National Laboratory)

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SUMMARY

The DYMOND code employs the ITHINK dynamic modeling platform to assess the 100-year dynamic evolution scenarios for postulated Global Nuclear Energy Parks. Firstly, DYMOND code has been developed by ANL(Argonne National Laboratory) to perform the fuel cycle analysis of LWR once-through and LWR-FBR mixed plant. Since the extensive application of DYMOND code has been requested, the first version of DYNOND has been modified to adapt the DUPIC, MSR and RTF fuel cycle.

DYMOND code is composed of three parts; the source language platform, input supply and output. But those platforms are not clearly distinguished. This report described all the equations which were modeled in the modified DYMOND code (which is called as DYMOND-DUPIC version). It divided into five parts;

Part A deals Model in Reactor History which is included amount of the requested fuels and spent fuels. Part B aims to describe Model of Fuel Cycle about fuel flow from the beginning to the end of fuel cycle. Part C is for Model in Re-processing which is included recovery of burned uranium, plutonium, minor actinide and fission product as well as the amount of spent fuels in storage and disposal. Part D is for Model in Other Fuel Cycle which is considered the thorium fuel cycle for MSR and RTF reactor. Part E is for Model in Economics. This part gives all the information of cost such as uranium mining cost, reactor operating cost, fuel cost etc..

I. INTRODUCTION

The DYMOND code employs the ITHINK dynamic modeling platform to assess the 100-year dynamic evolution scenarios for postulated Global Nuclear Energy Parks. The first DYMOND code was developed to apply LWR once-through and LWR-FBR mixed plant. Since the extensive application of DYMOND code has been requested, the first version of DYNOND has been modified in order to adapt the DUPIC, MSR and RTF fuel cycle.

DYMOND code is composed of three parts; the source language platform, input supply and output. But those platforms are not clearly distinguished.

This report described all the equations which were modeled in the modified DYMOND code. It divided into five parts;

Part A deals Model in Reactor History which is included amount of the requested fuels and spent fuels. Part B aims to describe Model of Fuel Cycle about fuel flow from the beginning to the end of fuel cycle. Part C is for Model in Re-processing which is included recovery of burned uranium, plutonium, minor actinide and fission product as well as the amount of spent fuels in storage and disposal. Part D is for Model in Other Fuel Cycle which is considered the thorium fuel cycle for MSR and RTF reactor. Part E is for Model in Economics. This part gives all the information of cost such as uranium mining cost, reactor operating cost, fuel cost etc..

Before describing model of DYMOND code, time which was used in this report should be defined to make reader better understanding.

- Definition of Time

The kinds of time are defined, which were used in this report.

1. Present time : T
2. Differential time to be considered : DT
3. Reactor construction time : $T_{Rx,const}$
4. Reactor licensing time : $T_{Rx,licen}$
5. Reactor pre-operational time : T_{preop} (Construction time plus Licensing time)
6. Reactor life time : $T_{Rx,life}$
7. Remaining reactor life time : $T_{Rx,lift,rem}$
8. Fuel enrichment time : T_{enrich}
9. Fuel fabrication time : $T_{fuel,fab}$
10. Fuel actual fabrication time : $T_{actual,fab}$
11. Fuel preparation time : $T_{fuel,prep}$
12. Fuel reprocessing time : $T_{fuel,rep}$

13. Starting time of reprocessing : $T_{start,reproc}$
14. Building time of DUPC fabrication plant : $T_{DUPIC,fab,build}$
15. Interim storage time of spent fuel : $T_{SF,sto}$
16. Spent fuel reprocessing time : T_{repro}
17. Building time of reprocessing plant : $T_{reproc,build}$

II. MODEL DESCRIPTION

A. MODELS IN REACTOR HISTORY

A.1 Reactors to Be Built

1. Number of Reactor to be Built

The number of reactors to be built can be expressed by the total reactor power needed, the capacity percent of i-type reactor and each reactor power. It results in,

$$N_{Rx,tot} = \sum_i \frac{(C_i / 100) \cdot P_{total,need}}{P_i},$$

where, C_i is the capacity percent of i-type reactor to total reactor capacity. $P_{total,need}$ is total reactor power needed and P_i is power of i-type reactors such as reactor under licensing, reactors under construction, fresh reactor, etc..

2. Total Reactor Power Needs

Total reactor power needs can be modeled with the summation of potential power of all the reactor and energy demand, that is,

$$MIN\left[\left(E_{demand,prediction} - \sum_i P_{potential,i}\right), E_{\max,const,rate}\right]$$

where, subscript, i represents reactor. This equation means the minimum energy needed, comparing the predicted energy demand subtracted the present total potential power with the maximum consumption rate of energy at time regarded.

3. Total Potential Power

Total potential power of reactors is each reactor power multiplied by the number of fresh reactors, reactors under construction, reactors under license, ready reactors and reactors to be needed fuels. It results in,

$$P_{potential,i} = N_i \cdot P_i$$

where, P_i is the i-type reactor power. Since the operating reactors represent the fresh reactors, reactor near retirement and reactors near shutdown, total power of operating reactors is the summation of operating reactors.

4. Maximum Construction Rate of Energy

Maximum construction rate of energy is the product of reactor energy at the rate of maximum construction of reactor and differential time considered,

$$E_{\max,const,rate} = f_{Rx,const,rate} \cdot C_{Rx,total,deployed} \cdot DT$$

where, $C_{Rx,total,deployed}$ is the total capacity of deployed reactors. Also, $f_{const,rate}$ is fraction of construction rate of reactors and it is set to 0.10 in present model.

5. Total Capacity of Deployed Reactor

It can be determined by number of reactors and each reactor power,

$$C_{Rx,total,deployed} = \sum_i N_{Rx,op,i} \cdot P_i$$

where, $N_{op,i}$ is the number of i-type reactor being operated such as the fresh reactors, reactor near retirement and reactors near shutdown.

6. Energy Demand

Energy demand can be predicted by energy growth rate, pre-operational time of reactor and differential time (DT). For example, energy demand of USA from 2000 to 2010 is assumed as a constant of 100 GWe (E_{2000}) and after 2010, the growth rate of energy demand assumed to be 2 % increase (r). And pre-operation time is summed the construction time and licensing time of reactor. The energy demand can be predicted and it is,

$$E_{demand,USA} = E_{2000} \cdot (1+r)^{T_{preop}-2010}$$

7. Energy Demand Prediction

Generally, the number of building reactors can be decided by energy demand prediction. Considering the USA energy demand as an example, the energy demand prediction is,

$$E_{demand,pred,USA} = E_{2000} \cdot (1+r)^{T+T_{preop}+2DT-2000}$$

The difference between energy demand and energy demand prediction is the consideration of pre-operation time of reactor and double of differential time considered. And pre-operation time is the reactor construction time plus the reactor licensing time.

8. Energy Demand Met

It is important to know how much energy demand met by the present energy source. The energy demand met by present energy source is,

$$E_{demand,met,i}(\%) = C_{Rx,deployed,i} \big/ E_{demand} \cdot 100$$

This equation shows the percent of demand met by each reactor type.

9. Deployed Reactor Capacity

The deployed reactor capacity is the product of the number of operating reactor and reactor power. It results in,

$$C_{Rx,deployed,tot} = \sum_i N_{Rx,op,i} \cdot P_i$$

where, subscript, i is reactor type.

10. Fraction of LWR-MOX Reactor

The fraction of LWR-MOX reactor compared to LWR reactor is,

$$MOX\_PLANT(\%) = N_{LWR-MOX,op} / \left(N_{LWR,op} + N_{LWR-MOX,op}\right) \cdot 100$$

11. Order of DUPIC Fabrication Plant

Considering DUPIC process, it should be ordered before starting the DUPIC fuel fabrication. If the prediction capacity of DUPIC fabrication plant is greater than the present capacity of DUPIC plant, the total capacity of DUPIC fabrication plant results in,

$$C_{DUPIC,fab,tot} = \left(N_{DUPIC,fab} + N_{DUPIC,build}\right) \cdot C_{DUPIC,fab}$$

where, DUPIC fabrication capacity, $C_{DUPIC,fab}$ is set to 0.4 {kt/y}, and building time of DUPIC plant is required 7 years in the present model. The prediction DUPIC fabrication plant is related to the energy supply plan. It results in

$$N_{predict,DUPIC} = \begin{bmatrix} \{1 - SWITCH(T,2050)\} \cdot \\ FORCST\left(F_{req,DUPIC}, 5, T_{DUPIC,fab,build} + DT\right) + \\ SWITCH(T,2050) \cdot \\ FORCST\begin{Bmatrix} SMTHN\left(F_{req,DUPIC}, 2 \cdot DT, 1\right), 10, \\ T_{DUPIC,fab,build} + DT \end{Bmatrix} \end{bmatrix}$$

And, the requirement of DUPIC fuel fabrication is

$$F_{req,DUPIC} = \sum_j F_{sotred,LWR,j} \quad \text{if } T < 2025$$

$$F_{req,DUPIC} = \sum_j F_{req,DUPIC,j} \quad \text{if } T \geq 2025$$

Total capacity of DUPIC fabrication plant is

$$C_{DUPIC,fab,tot} = N_{DUPIC,tot} \cdot C_{DUPIC,fab}$$

A.2 Reactors to Be Started

1. Reactor to be Started

In order to determine the number of reactors to be started, the amount of the available fuels should be determined early. The amount of available fuels can be expressed by the fuel amounts to be ready for reactors subtracted operating reactors multiplied by fuel consumption rate multiplied by time interval. Hence, the reactors to be started is

$$N_{Rx,start,i} = MIN\begin{bmatrix} MAX\left\{F_{avail,i,j} / \left(F_{ini,i,j} + C_{fuel,consum,rate,i,j}\right) \cdot DT, 0\right\}, \\ 0 * N_{Rx,ready,i} \end{bmatrix} + N_{Rx,ready,i}$$

where, i is reactor type and j is core type and $N_{Rx,ready,i}$ is the number of i-type reactor ready to operate. Here, the first term in the above bracket stands for the reactors to be started. Then the minimum number of reactors will be started, comparing the reactors to be ready with the available reactors which the available fuels can be supplied to. And the number of ready reactors represents the number of the reactors which are just before operation, that is, there is time delay of *DT* before reactor operation.

2. Available Fuels For Reactor

The available fuels for the reactor is

$$F_{avail,i,j} = F_{ready,i,j} - N_{Rx,op,i} \cdot C_{fuel,consum,rate,i,j} \cdot DT$$

where, $F_{ready,i,j}$ is the ready fuels to load into i-type reactor and j-type zone such as core, axial blanket (AB), radial blanket (RB) or inner blanket (IB). And $C_{fuel,consum,rate,i,j}$ is the capacity of the fuel consumption rate of i-type reactor and j-zone. The amount of initial fuel load, fuel consumption rate will be discussed in next section.

3. Operating Reactor

The operating reactors include the fresh reactors, reactors near retirement and reactor near shutdown. Hence,

$$N_{Rx,op,tot} = \sum_i N_{Rx,op,i}$$

where, i stands for fresh, near retirement and near shutdown reactors.

4. Thermal Power of Reactor

Thermal power can be calculated using efficiency,

$$P_{th,i,j} = \frac{P_i}{\eta_i} \cdot f_{heat,i,j}$$

where, P_i and η_i are reactor power and thermal efficiency of i-type reactor which will be given as an input. And, $f_{heat,i,j}$ is heat fraction from core, AB, RB and IB of i-type reactor to the total heat from a reactor.

5. Heat Fraction From Zone

Total heat of each reactor comes from core, AB, RB and IB. It results in,

$$f_{heat,i,X} = 1 - \sum_j f_{i,j}$$

where, i is reactor type such LWR, FBR etc. and j is zone type such as core, AB, RB and IB. If subscript, X is core, j's are AB, RB and IB except core. Further to AB, similarly, j's are core, RB and IB except AB.

6. Pre-operation Time of Reactor

Pre-operation time of reactor is the licensing time plus construction time of reactor.

$$T_{preop,i} = T_{licen,i} + T_{const,i}$$

A.3 Amount of Fuels Requested

1. Preparation Time of Fuel

Fuel preparation time is the summation of the enrichment and fabrication time,

$$T_{fuel,prep} = T_{enrich} + T_{fab}$$

A fuel is fabricated during the actual fuel fabrication time and differential time before fuel fabrication, hence, the actual fabrication time of a fuel is

$$T_{actual,fab} = T_{fab} - DT$$

2. Amount of Fuel Consumption

From the number of the reactors to be operated or ready, how much fuels will be consumed. That is,

$$F_{consum,i,j} = \left(F_{ini,i,j} + C_{fuel,consum,rate,i,j} \cdot DT\right) \cdot N_{Rx,i} + C_{fuel,consum,rate,i,j} \cdot DT \cdot N_{Rx,op,i}$$

3. Amount of Initial Fuel Load

The amount of the initial fuels for loading into reactor zone can be determined by the amount of fuel loading and fuel consumption rate, that is,

$$F_{ini,i,j} = F_{load,i,j} - C_{fuel,consum,rate,i,j} \cdot DT$$

where, $F_{load,i,j}$ is the amount of fuels loaded into i-type reactor and j-type zone.

4. Amount of Fuels loaded

The amount of fuels loaded into reactors can be calculated from the fuel consumption rate, fuel cycle length, number of batches and type of reactor core. It results in,

$$F_{load,i,j} = C_{fuel,consum,rate,i,j} \cdot L_{cycle,i,j} \cdot N_{batch,i,j} + F_{HTR,load} / 1000 \ \{\text{kt}\}$$

where, cycle length and number of batches are input data for each reactor type and $F_{HTR,load}$ is the amount of fuels loaded into HTR in Gt and calculated separately because of unknown the information for HTR except the total fuels loaded. It will be eliminated as soon as all the information of HTR are determined.

On the other hand, loading fuels are determined by the consumed fuels for each reactor. That is,

$$F_{load,i,j} = \left(F_{ini,i,j} + C_{fuel,consum,rate,i,j} \cdot DT\right) \cdot N_{Rx,start,i} + C_{fuel,consum,rate,i,j} \cdot DT \cdot N_{Rx,op,i}$$

5. Fuel Consumption Rate

The fuel consumption rate of each reactor and zone is a function of reactor thermal power, burnup and load factor,

$$C_{fuel,consum,rate,i,j} = \frac{P_{th,i,j} \cdot 365 \cdot C_{load,i,j}}{1000 \cdot Bu_{i,j}}, \ \{GWth*(d/yr)/(GWth\text{-}d/t)/(t/kt) = kt/yr\}$$

where, $P_{th,i,j}$, $C_{load,i,j}$ and $Bu_{i,j}$ are thermal power, load factor and burnup of i-type reactor and j-type zone, respectively.

6. Fuel Requested

Amount of fuels are generally requested during startup and refueling. That is,

$$F_{req,i,j} = F_{req,startup,i,j} + F_{req,refueling,i,j}$$

7. Fuel For Startup

Amount of fuel requested by startup is calculated by the number of reactors to need fuels and fuel amount for initial core load, that is,

$$F_{req,startup,i,j} = DELAY\left(N_{Rx,op,i} \cdot F_{ini,i,j} \cdot DT, DT, 0\right)$$

8. Fuel For Refueling

Amount of fuels requested by refueling can be determined by fuels for operating reactors, reactor near shutdown, reactor under construction, ready reactors. Therefore, the fuel requested by refueling is

$$F_{req,refueling,i,j} = DELAY\begin{bmatrix} \left(F_{op,i} - F_{near,shut,i} + F_{const,i} + F_{ready,i}\right) \cdot \\ C_{fuel,consum,rate,i,j} \cdot DT, DT \end{bmatrix}, \{kt\}$$

9. Each Type of Fuel Requested

From the fuels requested are used for determining the fuel amounts of UOX, metal fuel, ThO_2UO_2, MOX fuel, DUPIC fuel, MSR fuel, etc. The fuels requested for each type of reactor are calculated as followings;

$$F_{req,UOX,i,j} = f_{UOX,i,j} \cdot F_{req,i,j}\text{, \{kt\}} \qquad \text{for fuel requested by UOX}$$

$$F_{req,MOX,i,j} = f_{MOX,i,j} \cdot F_{req,i,j}\text{, \{kt\}} \qquad \text{for fuel requested by MOX}$$

$$F_{req,ThO_2UO_2,i,j} = f_{ThO_2UO_2,i,j} \cdot F_{req,i,j}\text{, \{kt\}} \qquad \text{for fuel requested by ThO2UO2}$$

$$F_{req,MSR,i,j} = f_{MSR,i,j} \cdot F_{req,i,j}\text{, \{kt\}} \qquad \text{for fuel requested by MSR}$$

$$F_{req,METAL,i,j} = f_{METAL,i,j} \cdot F_{req,i,j}\text{, \{kt\}} \qquad \text{for fuel requested by Metal}$$

$$F_{req,DUPIC,i,j} = f_{DUPIC,i,j} \cdot F_{req,i,j}\text{, \{kt\}} \qquad \text{for fuel requested by DUPIC}$$

where, the fractions of UOX is,

$$f_{UOX,i,j} = \left(1 - f_{MOX,i,j}\right) \cdot R_{frac,Th,i,j} \cdot \left(1 - f_{DUPIC,i,j}\right)$$

and $R_{ThO_2UO_2,frac,i,j}$ is thorium fraction factor and expressed by

$$R_{Th_2UO_2,,frac,i,j} = \left(1 - f_{METAL,i,j}\right) \cdot \left(1 - f_{Th_2UO_2,i,j}\right) \cdot \left(1 - f_{MSR,i,j}\right)$$

10. Total Fuel Fabrication Rate

There are several types of fuel considered, which are for LWR, LWR-MOX, RTF, MSR and LWR-DUPIC. Therefore, total fuel fabrication rates are the summation of the each fuel fabrication rates, that is,

$$F_{tot,fab,rate} = \sum_{i,j} F_{fab,rate,i,j}$$

where, subscripts, i and j are each reactor and zone.

11. Fuel Fabrication Rate

Fuel fabrication rate for each reactor and zone is the summation of each fuel type loaded in each reactor, that is,

$$F_{fab,rate,i,j} = \sum_{k} F_{fab,rate,i,j,k}$$

where, subscript, k stands for the fabrication rate for each fuel type such as LWR fuel (enriched U), MOX fuel, thorium type in RTF, MSR and DUPIC fuel type in LWR-DUPIC.

12. Fuel Fabrication Rate of Each Fuel Type

Fuel fabrication rate of each fuel type can be obtained,

$$F_{fab,rate,U,i,j} = M_{U,i,j} \Big/ \left(1 + \gamma_{fuel-to-tal,i,j}\right) \qquad \text{for enriched uranium}$$

$$F_{fab,rate,MOX,i,j} = DELAY\left(F_{req,MOX,i,j} \big/ DT, DT\right) \qquad \text{for MOX fuel}$$

$$F_{fab,rate,DUPIC,i,j} = DELAY\left(F_{req,DUPIC,i,j} \big/ DT, DT\right) \qquad \text{for DUPIC fuel}$$

$$F_{fab,rate,ThO_2UO_2,i,j} = F_{req,ThO_2UO_2,i,j} \cdot f_{ThO_2UO_2,i,j} \big/ DT \qquad \text{for thorium fuel}$$

where, $\gamma_{fuel-to-tal,i,j}$ is fuel-to-tail conversion ratio.

B. MODELS IN FUEL CYCLE

B.1 Uranium Resource

1. Uranium Mined

Uranium is mined for supplying the total amount of each fuel type to each reactors. Also, the uranium mined will be converted to fuel by the conversion ratio of uranium mined to fuel. The amount of uranium mining can be calculated as followings

$$M_{U,mined,i,j} = \gamma_{fuel-to-ore} \cdot \frac{\sum_k F_{req,i,j,k}}{DT}$$

where, $\gamma_{fuel-to-ore}$ is conversion ratio of fuel-to-ore and i, j and k represent reactor type, zone type such as core, axial blanket, radial blanket and inner blanket, and fuel type respectively. The fuel can be classified by Metal, UOX, ThO_2UO_2, MSR fuel etc..

2. Uranium in ThO_2UO_2

- Uranium needed for thorium cycle in RTF

$$M_{U,ThO_2UO,j} = F_{ThO_2UO_2,RTF,j} \cdot \left(1 - f_{ThO_2UO_2,RTF}\right)$$

where, $f_{ThO_2UO_2,RTF}$ is thorium fraction in Radkowsky Thorium Fuel Cycle and given as an input.

- Uranium needed for Non-Actinide MSR

$$M_{U,non-Act-MSR,j} = F_{non-Act-MSR,j} \cdot f_{non-Act-MSR,j}$$

where, $f_{non-Act-MSR,j}$ is uranium fraction of Non-Actinide MSR.

3. Depleted Uranium in MOX Fuel

It is important to know the fraction of depleted uranium in MOX fuel because of utilization of the uranium ore and recovered uranium fro spent fuels. The fraction of depleted uranium in MOX fuel can be calculated as followings;

$$f_{depU,MOX} = 1 - \left(f_{U,SF,MOX,j} + f_{Pu,SF,MOX,j} + f_{MA,SF,MOX,j} \right)$$

if $M_{U,SF,MOX} > M_{U,need,MOX}$

$$f_{depU,MOX} = 1 - \left(f_{Pu,SF,MOX,j} + f_{MA,SF,MOX,j} \right)$$

if $M_{U,SF,MOX} \le M_{U,need,MOX}$

$M_{U,need,MOX}$ and $M_{U,SF,MOX}$ are uranium needed in each reactor and zone and uranium in spent fuel, respectively. And these amounts are,

$$M_{U,need,MOX} = M_{req,MOX} \cdot f_{U,SF,MOX,j}$$

$$M_{U,SF,MOX} = M_{reproc,MOX,j} \cdot f_{U,SF,MOX,j} \cdot \left(1 - f_{loss,MOX,j}\right)$$

4. Uranium Ore Consumed And Remaining Resources

Uranium ore resources are composed of known resources, unknown resources, and imagine resources. From these resources, remaining of known resources, unknown resources and imagine resources could be found. The remaining of known resources is,

$$M_{U,known,res,rem} = MAX\left[M_{U,ore} - \left(M_{U,ini,res} - M_{U,known,res}\right), 0\right]$$

Similarly, the remaining of unknown resources is

$$M_{U,unknown,res,rem} = MAX\left[M_{U,ore} - M_{U,ini,res,rem} - \begin{pmatrix} M_{U,ini,res} - M_{U,unknown,res} \\ - M_{U,known,res} \end{pmatrix}, 0 \right]$$

Also, the remaining of imagine resources is

$$M_{U,imagine,res,rem} = MAX\begin{bmatrix} M_{U,ore} - M_{U,known,res,rem} - M_{U,unknown,res,rem} - \\ \left(M_{U,ini,res} - M_{U,unknown,res} - M_{U,known,res} - M_{U,imagine,res}\right), 0 \end{bmatrix}$$

where, $M_{U,ini,res}$ is initial resources and it is determined by

$$M_{U,ini,res} = \begin{bmatrix} \left(M_{U,imagine,res} + M_{U,unknown,res} + M_{U,known,res}\right) \cdot C_{known,unknown,imagine} + \\ + \left(M_{U,known,res} + M_{U,unknown,res}\right) \cdot C_{known,unknown} + M_{U,known,res} \cdot C_{known,res} \end{bmatrix}$$

where, C_X is a considering factor for known, known-unknown and known-unknown-imagine. From these relationship, new discovery of uranium, $M_{U,new,dis}$ can be found,

$$M_{U,new,dis} = M_{U,imagine,res} - M_{U,imagine,res,rem}$$

Also, total uranium resources remaining, $M_{U,tot,res,rem}$ can be found,

$$M_{U,tot,res,rem} = M_{U,known,res,rem} + M_{U,unkonwn,res,rem}$$

5. Remaining Ore Need

It is needed to know how much ore is remained. The amount of ore needed is a function of life time and ore consumption rate of each reactor. That is,

$$M_{U,ore,need,i,j} = T_{Rx,life,rem,i} \cdot C_{U,ore,consum,rate,i,j}$$

where, $C_{U,ore,consum,rate,i,j}$ is ore consumption rate of each reactor and zone, and

$$C_{U,ore,consum,rate,i,j} = C_{fuel,consum,rate,i,j} \cdot \gamma_{fuel-to-ore,i,j}$$

Also, $\gamma_{fuel-to-ore,i,j}$ is fuel to ore conversion ratio of each reactor and zone.

6. Remaining Life Time of Reactor

The remaining life time of reactor, $T_{Rx,life,rem,i}$ is calculated, considering reactor life time and the rate of reactor construction which is determined by reactor construction time and fuel preparation time.

$$T_{Rx,life,rem,i} = \begin{bmatrix} N_{Rx,fresh,i} \cdot \left\{\left(T_{Rx,life,i} + T_{Rx,preop,i}\right)/2 - T_{fuel,prep,i}\right\} + \\ N_{Rx,underconst,need,fuel} \cdot \left(T_{Rx,life,i} - T_{fuel,prep,i}/2\right) \end{bmatrix}$$

where, $T_{Rx,preop,i}$, $T_{fuel,prep,i}$ and $N_{Rx,underconst,need,fuel}$ are pre-operation time for reactor, fuel preparation time and the number of reactors under construction needed fuel.

7. Separative Work Unit

In the enrichment, three streams of materials are exist, the input (natural uranium), output (enriched uranium) and residue or tails (depleted uranium). If M_F is the mass of uranium in the feed material supplied to the separation cascade, M_P is the mass of product withdrawn, and M_W is the mass of the waste, a uranium mass balance requires that

$$M_F = M_P + M_W$$

assuming, as is generally true, that there is no appreciable loss of uranium during the operation. A similar balance can be applied to the uranium-235 only; thus,

$$M_F \cdot x_F = M_P x_P + M_W x_W$$

where, x_F , x_P and x_W are the assays in feed, product and waste, respectively. By eliminating M_W from these two equations, the result is

$$\frac{M_F}{M_P} = \frac{x_P - x_W}{x_F - x_W}$$

This equation gives the mass of uranium feed of assay x_F required per unit mass of uranium product of assay x_P , assuming a tail assay of x_W .

The cost of enrichment is determined by the amount of the work that has to be done to achieve the enrichment. A so-called value function has been developed on the basis of the theory of the gaseous-diffusion cascade. It is represented by

$$V(x) = (1 - 2x)\ln\frac{1 - x}{x}$$

Because x is a fraction, the value function, which is characteristic of a given assay, is a fraction and has no units. It is used to determine the work required to yield a product of a desired assay from a given feed with a specified waste.

The effort expended in separating a mass M_F of feed of assay x_F into a mass M_P of product of assay x_P and waste of mass M_W and assay x_W is expressed in terms of the number of separative work units (SWU) needed. This given in terms of the respective value functions by

$$SWU = M_W \cdot V(x_W) + M_P \cdot V(x_P) - M_F \cdot V(x_F)$$

Since the value functions have no units, the SWU will have the same units as the masses M_F, M_P, and M_W. The general practice is to state the number of seperative work units in terms of kilograms of uranium. Upon dividing the equation through by M_P, the result

$$\frac{SWU}{M_P} = \frac{M_W}{M_P} \cdot V(x_W) + V(x_P) - \frac{M_F}{M_P} \cdot V(x_F)$$

gives the number of separative work units received per unit mass of product.

Let us apply to each reactor and zone. The value functions are,

$$V_{fuel,i,j}\left(x_{fuel,enrich,i,j}\right) = \left(1 - 2x_{fuel,enrich,i,j}/100\right)\ln\frac{100 - x_{fuel,enrich,i,j}/100}{x_{fuel,enrich,i,j}/100}$$

$$V_{nat}\left(x_{nat,enrich}\right) = \left(1 - 2x_{nat,enrich}/100\right)\ln\frac{100 - x_{nat,enrich}/100}{x_{nat,enrich}/100}$$

$$V_{tail}\left(x_{tail,enrich}\right) = \left(1 - 2x_{tail,enrich}/100\right)\ln\frac{100 - x_{tail,enrich}/100}{x_{tail,enrich}/100}$$

And SWU is,

$$\frac{SWU_{i,j}}{M_{fuel,i,j}} = \frac{M_{tail}}{M_{fuel,i,j}} \cdot V_{tail}\left(x_{tail,enrich}\right) + V_{fuel,i,j}\left(x_{fuel,enrich,i,j}\right) - \frac{M_{nat,enrich}}{M_{fuel,i,j}} \cdot V_{nat,enrich}\left(x_{nat,enrich}\right)$$

Fuel to ore conversion ratio is,

$$\gamma_{fuel-to-ore,i,j}\left(\equiv \frac{M_{ore}}{M_{fuel,enrich,i,j}}\right) = \frac{x_{fuel,enrich,i,j} - x_{tail,enrich}}{x_{nat,enrich} - x_{tail,enrich}}$$

Fuel to tail conversion ratio is,

$$\gamma_{fuel-to-tail,i,j}\left(\equiv\frac{M_{tail}}{M_{fuel,enrich,i,j}}\right)=\gamma_{fuel-to-tail,i,j}-1$$

Enrichment rate is,

$$R_{U,enrich,rate,i,j}=M_{U,mined,i,j}\cdot SWU_{i,j}/\gamma_{fuel-to-ore,i,j}$$

Hence, total enrichment rate is

$$R_{U,tot,enrich,rate}=\sum_{i,j}M_{U,mined,i,j}\cdot SWU_{i,j}/\gamma_{fuel-to-ore,i,j}$$

where, i and j stand for reactor type and zone type.

B.2 Spent Fuels

1. Spent Fuel Production

Spent fuel will be discharged after burning in the reactor. The spent fuel production is defined as an initial fuel loaded and fuel consumption rate;

$$F_{SF,i,j}=DELAY\left(C_{fuel,consum,rate,i,j}\cdot N_{Rx,op,i}\cdot DT,DT\right)+F_{ini,i,j}\cdot N_{Rx,shutdown,i}\cdot DT$$

where, DELAY is built-in function in ITHINK program and subscripts, i and j stand for reactor and zone, respectively.

2. Spent Fuel Inventory

Total spent fuel inventories are composed of Metal uranium fuels, MOX fuels, UOX fuels, Thorium fuels and DUPIC fuels. Hence,

$$M_{SF,total}=M_{SF,MetalU,tot}+M_{SF,MOX,tot}+M_{SF,UOX,tot}+M_{SF,ThO_2UO_2,tot}+M_{SF,DUPIC,tot}$$

where, each total mass of materials is

$$M_{SF,X,tot}=\sum_{i,j}M_{SF,X,repos,i,j}+\sum_{i,j}M_{SF,X,i,j}$$

$$M_{SF,X,repos,i,j} = M_{SF,X,repos,i,j} \cdot f_{SF,X,i,j}$$

$$M_{SF,X,i,j} = M_{SF,X,i,j} \cdot f_{SF,X,i,j}$$

where, subscript, X stands for Metal uranium fuels, MOX fuels, UOX, fuels, Thorium fuels and DUPIC fuels, etc..

3. Spent Fuel to Repository

In case of LWRs, no reprocessing of fuel is required if there is no MOX program. When the MOX or FBR programs are started, all spent fuels will be transferred to the reprocessing processes. But there is time delay which is in interim storage before reprocessing. Therefore, it should be considered whether LWR MOX or FBR plants will be constructed or not before determining the amount of spent fuel to repository. Here, both cases are considered as following;

$$\text{"To reprocess"} = \begin{cases} 0, \text{if TIME} < \text{"Start reprocess"} \\ \text{"Reprocessing capacity"} \quad \text{otherwise} \end{cases},$$

This means that all the spent fuels in the interim storage inventories will be consumed after time of “Start reprocess”, otherwise, all the spent fuels in the interim storage inventories go to the permanent disposal. For example,

For LWR-DUPIC case,

$$F_{SF,repos,LWR-DUPIC} = \begin{bmatrix} SWITCH\left(T_{SF,sto,LWR}, T_{present} - T_{start,reproc}\right) \cdot \\ INIT\left(F_{SF,LWR} / T_{SF,sto}\right) \end{bmatrix}$$

For FBR case,

$$F_{SF,repos,FBR} = \left(1 - T_{reproc,FBR}\right) \cdot \begin{bmatrix} DELAY\left(F_{SF,FBR,j}, T_{SF,sto}, 0\right) + \\ SWITCH\left(T_{SF,sto}, T - T_{start,reproc}\right) \cdot \\ INIT\left(F_{SF,FBR,j}\right) / T_{SF,sto} \end{bmatrix}$$

where, “Start reprocess” is the given beginning date of spent fuel reprocessing for every type of reactor, and $T_{reproc,i}$

$$T_{reproc,i} = SWITCH\left(T_{stop}, T_{reproc,start}\right)$$

C. MODELS IN REPROCESSING

C.1 Reprocessing Plant

1. Reprocessing Capacity

Reprocessing plant capacity should be known in order to know how much spent fuels are reprocessing or reprocessed. The reprocessing capacity is determined by the reprocessing capacity demand required by each reactor, the capacity of reprocessing plant, number of reprocessing plants and start time for reprocessing the spent fuels for each reactor which is existed or to be built. The reprocessing capacity is

$$C_{reproc} = C_{plant,reproc} \cdot N_{plant,reproc}$$

where, $C_{plant,reproc}$ is the capacity of reprocessing plant. And if the energy by MSR is not demanded, reprocessing capacity is equal to the product of the number of reprocessing plants and capacity of the reprocessing plant. Otherwise, no reprocessing capacity is needed before 2045. And in-between 2045 and 2065 one reprocessing plant will be built and until 2085, two reprocessing plants will be built. The capacity of all the reprocessing plants are set to 14 {kt/yr} except the 1000 {kt/yr} for the capacity of reprocessing plant of FBR.

2. Reprocessing Plant

In order to reprocess the spent fuels, it should be known how many reprocessing plant will be ordered. If the prediction of the number of reprocessing plant is greater than that of the existed reprocessing plant and reprocessing under construction, the capacity of reprocessing plant sets to be 1. Otherwise, that set to be 0. It means that one reprocessing plant will be built if the present reprocessing plants included with the plant under construction is less than the necessary plant. Here, the reprocessing plant can be predicted as followings;

$$N_{rep,predicted} = \begin{bmatrix} \{1 - SWITCH(T,2060)\} \cdot \\ FORCST\{F_{SF,sotred,i},5 + SWITCH(T,2012) \cdot 5, T_{reproc,build} + DT\} + \\ FORCST\{F_{SF,sotred,i},5 + SWITCH(T,2012) \cdot 5, T_{repproc,build} + DT\} + \\ \{SWITCH(T,2060)\} \cdot \\ FORCST\{SMTHN(F_{SF,sotred,i},2 \cdot DT,1),10, T_{reproc,build} + DT\} \end{bmatrix}$$

where, the building time of reprocessing plant is set to 4 years. And, the request of the reprocess plant building is,

$$N_{req,reproc,build} = DELAY(N_{reproc,predicted}, T_{reproc,build} + DT)$$

3. DUPIC Reprocessing Capacity

DUPIC reprocessing capacity can be determined by DUPIC fabrication plants and the capacity of DUPIC fabrication plants similar to reprocessing plant capacity of other reactors. That is,

$$C_{DUPIC,rep} = C_{DUPIC,plant,reproc} \cdot N_{DUPIC,plant,reproc}$$

But the differences of DUPIC reprocessing plant from other reprocessing plant is the capacity of reprocessing plant and it is set to 0.4 {kt/(yr.Rx)}.

C.2 Dispensation of Reprocessed Materials

The reprocessing cycles for MOX, MSR, FBR and DUPIC are considered. During reprocessing the spent fuels, four kinds of materials can be recovered such as burned Uranium(U), Plutonium(Pu), Minor Actinide(MA) and Fission Products(FP). In MOX case, Pu, U and MA recovered from the spent fuels as well as depleted uranium(DU) will be put into MOX fuel. In MSR case, DU and thorium(Th) will be put into MSR fuel as well as Pu and MA recovered from the spent fuels. In case of DUPIC reprocess, all materials included FP are not separated during the reprocess except the gaseous FP or volatile materials. It guaranteed highly proliferation resistances of DUPIC process. Now, considering how much materials are existed in high level waste disposal, the dispensation of each spent fuel can be calculated. At first, Total amounts of each material in spent fuel are calculated by fraction of each material such as U, Pu, MA and FPs.

1. Burned Uranium in Spent Fuel

   Burned uranium in spent fuel is

$$M_U = \left[\sum_i \sum_j \left(M_{SF,i,j} \cdot f_{U,i,j} \right) \right]$$

where, i and j stand for reactor type and zone respectively. And, $M_{SF,i,j}$ is the amount of total spent fuels.

2. Plutonium in Spent Fuel

   Plutonium in spent fuel is

$$M_{Pu} = \left[\sum_i \sum_j \left(M_{SF,i,j} \cdot f_{Pu,i,j} \right) \right]$$

3. Minor Actinides In Spent Fuel

   Minor actinides in spent fuel is

$$M_{MA} = \left[\sum_i \sum_j \left(M_{SF,i,j} \cdot f_{MA,i,j} \right) \right]$$

4. Fission Product In Spent Fuel

   Fission product in spent fuel is

$$M_{FP} = \sum_i \sum_j \left\{ M_{SF,i,j} \cdot \left(1 - f_{U,i,j} - f_{Pu,i,j} - f_{MA,i,j} \right) \right\}$$

C.3 Materials Recovered by Reprocessing Plant

In order to know the material amount recovered by reprocessing plant, M_X should be total materials for reprocessing, M_X 's except M_{FP} multiplied by fraction of loss ($f_{loss,X}$) during recovering the materials. It will be sent to high level waste disposal. The subscript “X” stands for the U, Pu, MA and FP. Each reprocessing loss ($f_{loss,X}$) of U, Pu and MA set to be 0.2 % during the reprocessing.

1. Recovered Material During Reprocessing

The spent fuel will be reprocessed if LWR-MOX or FBR plants are deployed. Here, the recovered materials during reprocessing will be found.

- Burned U in MOX

$$M_{U,MOX} = M_{req,MOX} \cdot f_{SF,MOX} / DT$$

- MA in MOX

$$M_{MA,MOX} = M_{req,MOX} \cdot f_{SF,MA} / DT$$

- Pu in MOX

$$M_{Pu,MOX} = M_{req,Pu} \cdot f_{SF,Pu} / DT$$

2. Waste in Disposal

Now, it is important to know how much wastes of the spent fuel in disposal are existed. The wastes of the spent fuel in disposal are dependent on the type of reprocessing technologies such as PUREX, APUREX, PYRO. The amount of wastes in spent fuel disposal, hence, can be determined as followings;

$M_{waste,disp,i} = 1 - \left(f_{Pu,i,j} + f_{Ui,j}\right) \cdot \left(1 - f_{loss,X}\right)$ for PUREX process

Since PUREX process is the object of recovering Pu and U only for using to MOX fuel, this equation states that the waste amount using PUREX process is the remains of spent fuels after recovered Pu and burned U, or MA, FPs and loss of Pu and burned U during reprocess. In case of using the APUREX or PYRO process, Pu, U and MA are usually recovered. Hence, the amount of waste in disposal is

$M_{waste,disp,i} = 1 - \left(f_{Pu,i,j} + f_{U,i,j} + f_{MA,i,j}\right) \cdot \left(1 - f_{loss,X}\right)$ for APUREX or PYRO process

3. Plutonium Availability

Now, let’s consider availability of plutonium. Pu amount for startup the reactor and refueling, for new reactors and unused Pu.

The Pu amount for reactor startup is the product of the fuel amount for startup, MOX fraction for each reactor and zone and Pu fraction in MOX of each reactor and zone. It results in

$$M_{Pu,start} = \left[\sum_i \sum_j \left(F_{start,i,j} \cdot f_{MOX,i,j} \cdot f_{Pu,i,j} \right) \right]$$

Plutonium amount for refueling can be calculated as followings;

$$M_{Pu,refueling} = \left[\sum_i \sum_j \begin{pmatrix} F_{refueling,i,j} - DELAY\left(N_{underconst,i} \cdot C_{fuel,consum,rate} \cdot DT, DT, 0 \right) \cdot \\ f_{MOX,i,j} \cdot f_{Pu,i,j} \end{pmatrix} \right]$$

Plutonium amount for new reactors can be calculated as followings;

$$M_{Pu,new} = M_{Pu,re\mathrm{cov}ed} - M_{Pu,refueling}$$

Plutonium amount unused can be calculated as followings;

$$M_{Pu,unused} = M_{Pu,re\mathrm{cov}ed} - M_{Pu,refueling} - M_{Pu,start}$$

D. MODELS IN OTHER FUEL CYCLE

D.1 Thorium Cycle

1. Thorium Consumption

Thorium is supposed to be used for RTF (Radkowsky) and MSR. Total amounts of thorium consumption will be divided into ore consumption in RTF and MSR.

- Thorium to RFT

The amount of thorium to RFT is determined in terms of thorium fuel requested and thorium fraction in RTF reactor. It results in

$$F_{ThO_2UO_2,RTF,j} = M_{ThO_2UO_2,req,j} \cdot f_{ThO_2UO_2,RTF,j} / DT$$

- Thorium to Non-Actinide MSR

Similar to RTF, the amount of thorium to Non-Actinide MSR is

$$F_{ThO_2UO_2,Non-Act-MSR,j} = M_{ThO_2UO_2,Non-Act-MSR,req,j} \cdot f_{ThO_2UO_2,Non-Act-MSR,j} / DT$$

- Thorium to Actinide MSR

Similarly, the amount of thorium to Actinide MSR is

$$F_{ThO_2UO_2,Act-MSR,j} = M_{ThO_2UO_2,Act-MSR,req,j} \cdot f_{ThO_2UO_2,Act-MSR,j} / DT$$

2. Thorium to MSR Operation

Thorium amount for MSR operation is needed as followings;

$$F_{ThO_2UO_2,MSR,op,j} = M_{ThO_2UO_2,MSR,op,j} \cdot N_{MSR,op}$$

3. Thorium Fuel Flow to MSR

Thorium is generally used for MSR reactors which are classified with two kinds of MSR reactors such as Non-Actinide MSR reactor and Actinide MSR reactor. Each MSR reactor has two options of loading the initial fuels into reactor core as ^{235}U and Pu and MA which are recycled through the reprocessing.

- Startup Fuel in MSR

$$F_{start,MSR} = F_{ThO_2UO_2,MSR,recy,j} + F_{ThO_2UO_2,Act-MSR,j} + F_{ThO_2UO_2,Non-Act-MSR,j}$$

where,

$$F_{ThO_2UO_2,MSR,recy,j} = N_{Rx,shutdown,i} \cdot F_{ini,load,i,j}$$

$$F_{ThO_2UO_2,Act-MSR,j} = DELAY\left(F_{ThO_2UO_2,req,Act-MSR,j} / DT , DT\right)$$

$$F_{ThO_2UO_2,Non-Act-MSR,j} = M_{U,enrich,Non-Act-MSR,j} + F_{ThO_2UO_2,Non-Act-MSR,rate,j}$$

$$M_{U,enrich,Non-Act-MSR,j} = M_{MSR,i} / \left(1 + \gamma_{fuel-to-tail,MSR,j}\right)$$

$$F_{ThO_2UO_2,Non-Act-MSR,rate,j} = F_{Non-Act-MSR,req,j} \cdot f_{ThO_2UO_2,Non-Act-MSR,j} / DT$$

- Fuel Request for MSR

There are two options which use the initial fuel loaded core as ^{235}U or Pu and MA. Fuels requested for both cases can be expressed by,

$$F_{ThO_2UO_2,MSR,req,j} = F_{req,i,j} \cdot f_{ThO_2UO_2,MSR,j} \cdot \left(1 - Option_{MSR}\right)$$

where, $Option_{MSR}$ is option number which is equal to 0 or 1 for fuel type of initial load fuels such as ^{235}U and recycled Pu and MA, respectively. Also, before starting recycle for MSR, $Option_{MSR}$ is equal to 0. Otherwise, $Option_{MSR}$ has non zero value.

4. Mined Thorium Ore

Since thorium can be used to MSR reactors or RTF reactor, thorium will be mined from ore before starting MSR or RFT reactor and the amount of thorium will be,

$$F_{ThO2UO2,mined,j} = \left(F_{ThO_2UO_2,Non-Act-MSR,j} + F_{ThO_2UO_2,Act-MSR,j} + F_{ThO_2UO_2,MSR,op,j} + F_{ThO_2UO_2,RTF,j}\right)$$

where, subscript j indicates the zone such as core, AB, IB and RB. And all parameters in right hand side of the above equation were found before.

5. Thorium Fuel Feed to MSR

In MSR reactor, depleted uranium as well as thorium will be fed to complete the thorium cycle. The amount of fuel feed to MSR can be expressed by

$$F_{feed,MSR,j} = F_{DU,feed,MSR,op,j} + F_{ThO_2UO_2,feed,MSR,op,j}$$

where,

$$F_{DU,feed,MSR,op,j} = \left(F_{DU,feed,Non-Act-MSR,op,j} \text{ or } F_{DU,feed,Act-MSR,op,j}\right) \cdot N_{Rx,op,MSR}$$

$$F_{ThO_2UO_2,feed,MSR,op,j} = \left(F_{ThO_2UO_2,feed,Non-Act-MSR,op,j} \text{ or } F_{ThO_2UO_2,feed,Act-MSR,op,j}\right) \cdot N_{Rx,op,MSR}$$

6. Depleted Uranium Feed to MSR

In MSR reactor, it is necessary to know depleted uranium feed to MSR. The amount of depleted uranium is,

$$F_{DU,feed,MSR} = \sum_j F_{DU,MSR,op,j} + \sum_j F_{DU,Non-Act-MSR,j}$$

where, subscript j indicates each zone of a reactor. This equation means that the depleted uranium feed to MSR is divided into two groups, depleted uranium will be supplied to reactor operating as well as non-actinide MSR. And

$$F_{DU,MSR,op,j} = F_{feed,Non-Act-MSR,j} \cdot N_{op,i} \quad \text{for Non-Actinide MSR}$$

$$F_{DU,MSR,op,j} = F_{feed,Act-MSR,j} \cdot N_{op,i} \quad \text{for Actinide MSR}$$

where, subscript i indicates fresh, near retirement and near shutdown reactors. Also, the depleted uranium feed to actinide MSR is

$$F_{DU,Act-MSR,j} = F_{req,Act-MSR,j} \cdot f_{DU,Act-MSR,j} / DT$$

D.2 Recycle of MSR

1. Recycled Fuel from MSR

It is important to know how much spent fuels should be recycled for MSR. The recycle fuels for MSR can be obtained by,

$$F_{recy,MSR,j} = N_{Rx,shdown,MSR,j} \cdot F_{ini,MSR,j}$$

2. Fission Product from MSR

$$F_{FP,MSR,j} = F_{feed,MSR,j} - F_{recy,loss,Pu-MA,MSR,j}$$

where, $F_{recy,loss,Pu-MA,MSR,j}$ is the wastes of Pu and MA from MSR spent fuels.

3. ^{233}U Fraction in Spent Fuel Storage and Repository

It is necessary to know the amount of ^{233}U in spent fuel and repository. It can be calculated by followings;

- ^{233}U in spent fuel

$$M_{U233,i} = F_{SF,i} \cdot f_{SF,U233,i,j}$$

- ^{233}U in repository

$$M_{U233,i,j} = \left(F_{SF,i,j} + F_{SF,repo,i,j}\right) \cdot f_{SF,U233,i,j}$$

where, fraction of ^{233}U is

$f_{SF,U233,i} = f_{SF,U233,Act-MSR,i}$ for Actinide MSR reactor

$f_{SF,U233,i} = f_{SF,U233,Non-Act-MSR,i}$ for Non-Actinide MSR reactor

4. MSR Fuel Processing

It is necessary to know the amount of uranium-233 exists in spent fuel and repository. It can be calculated by followings;

▪ Excess U-233

$$M_{U233,excess} = f_{excess,fm,pro-fuel} \cdot f_{excess,fm,U233}$$

where, subscript, *fm* and *pro-fuel* stand for fissile material and processed fuel, respectively. $f_{excess,fm,pro-fuel}$ and $f_{excess,fm,U233}$ is given as an input.

▪ Excess Fission Material

$$M_{excess,fm} = f_{excess,fm,pro-fuel} \cdot F_{rate,pro-fuel,MSR,j}$$

▪ Fuel Producing for MSR

$$F_{rate,pro-fuel,MSR,j} = F_{MSR,j} \cdot f_{rep,MSR,j} / DT$$

▪ Material Loss During Producing MSR Fuel

$$F_{loss,rate,pro-fuel,MSR,j} = F_{rate,pro-fuel,MSR,j} \cdot f_{loss,pro-fuel,MSR,j}$$

▪ Material Back to MSR

$$M_{back-to-MSR,MSR,j} = F_{rate,pro-fuel,MSR,j} - M_{excess,fm,MSR,j} - M_{waste,from-MSR,MSR,j} - F_{loss,rate,pro-fuel,MSR,j}$$

▪ Waste from MSR

$$M_{waste,from-MSR,MSR,j} = F_{rate,pro-fuel,MSR,j} \cdot f_{FP,fuel-pro,MSR,j}$$

E. MODELS IN ECONMICS

E.1 Mining and Enrichment Cost

Mining and enrichment costs includs the cost of mining, enrichment and conversion.

1. Total Mining Cost

$$Co_{mining,i,j} = M_{mined} \cdot Co_{mining,unit,i,j} \quad \{\text{M\$/y}\}$$

2. Enrichment Cost

$$Co_{enrich,i,j} = C_{enrich,rate,i,j} \cdot Co_{enrich,unit,i,j} \quad \{\text{M\$/y}\}$$

where, enrichment rate is

$$C_{enrich,rate,i,j} = M_{mined} \cdot SWU_{i,j} / \gamma_{fuel-to-ore,i,j} \quad \{\text{kt-SWU/yr}\}$$

3. Conversion Cost

$$Co_{conv,i,j} = \left(M_{mined,i,j} + M_{mined,Thorium,i,j}\right) \cdot Co_{conv,unit,i,j} \quad \{\text{M\$/y}\}$$

4. Total Cost of Each Process

Total cost for mining

$$Co_{tot,mining} = \sum_{i,j} \left(Co_{mining,i,j} + Co_{mining,ThO_2UO_2,i,j}\right) \Big/ C_{Rx,tot,deployed} \Big/ 8.76 \quad \{\text{M\$/y}\}$$

Total cost for conversion

$$Co_{tot,conv} = \sum_{i,j} Co_{conv,i,j} \Big/ C_{Rx,tot,deployed} \Big/ 8.76 \quad \{\text{M\$/TWe-h}\}$$

5. Total Cost of Mining and Enrichment

$$Co_{tot,mining-enrich} = \sum_k (Co_k) \text{ \{M\$/y\}}$$

where, subscript k stands for the mining, conversion and enrichment

E.2 Power Production Cost

To estimate the power production cost, cost of reactor operating and maintenance and capital cost are considered.

1. Reactor O&M Cost

$$Co_{Rx,tot,O\&M} = \sum_i N_{Rx,op,i} \cdot P_i \cdot f_{O\&M,inCC,i} \cdot Co_{Rx,CC,i} / 100$$

{R * GWe/R * %/y * M$/GWe *1/% = M$/y}

where, $f_{O\&M,inCC,i}$ and $Co_{Rx,CC,i}$ stand for fraction of O&M cost in capital cost and capital cost, respectively.

2. Reactor Capital Cost

$$Co_{Rx,tot,CC} = \sum_i N_{Rx,i} \cdot P_i \cdot Co_{Rx,CC,i} \quad \text{\{R/y * GWe/R * M\$/ GWe = M\$/y\}}$$

3. Power Production Cost

$$Co_{power,product} = Co_{Rx,tot,O\&M} + Co_{Rx,tot,CC} \quad \text{\{R/y * GWe/R * M\$/ GWe = M\$/y\}}$$

E.3 Fuel Fabrication Cost

To determine the fabrication cost of a fuel, the fabrication rate for each reactor and zone, total deployed reactor capacity, fuel fabrication unit cost are considered as followings;

$$Co_{fuel,fab} = C_{fuel,fab,recy,i,j} \cdot Co_{fuel,fab,unit,i,j} / C_{Rx,deployed} / 8.76 \text{ \{\$/TWeh\}}$$

E.4 Fuel Storage Cost

After burning the fuel materials, those will be transferred to the storage reservoir.

1. Spent Fuel Storage Cost

$$Co_{SF,sto,i,j} = \left(F_{SF,i,j} + F_{SF,sto,i,j}\right) \cdot Co_{SF,unit} + Co_{SF,sto,charge,i,j} \cdot F_{SF,prod,i,j} \ \{\text{M\$/y}\}$$

2. Total Spent Fuel Storage Cost

$$Co_{SF,sto,tot,i,j} = \sum_{i,j} Co_{SF,sto,i,j} \Big/ C_{Rx,tot,deployed} \Big/ 8.76 \ \{\text{M\$/y}\}$$

3. Minor Actinide Storage Cost

$$Co_{MA,sto,i,j} = M_{MA} \cdot Co_{MA,sto,unit} \Big/ C_{Rx,tot,deployed} \Big/ 8.76 \ \{\text{M\$/TWe-h}\}$$

4. Separated Plutonium Storage Cost

$$Co_{Pu,sto,i,j} = M_{Pu} \cdot Co_{Pu,sto,unit} \Big/ C_{Rx,tot,deployed} \Big/ 8.76 \ \{\text{M\$/TWe-h}\}$$

5. Depleted Uranium Storage Cost

$$Co_{depU,sto,i,j} = M_{depU} \cdot Co_{depU,sto,unit} \Big/ C_{Rx,tot,deployed} \Big/ 8.76 \ \{\text{M\$/TWe-h}\}$$

6. Burned Uranium Storage Cost

$$Co_{burnU,sto,i,j} = M_{burnU} \cdot Co_{burnU,sto,unit} \Big/ C_{Rx,tot,deployed} \Big/ 8.76 \ \{\text{M\$/TWe-h}\}$$

7. Total Storage Cost

$$Co_{tot,sto} = \sum_{X}\sum_{i,j} Co_{X,sto,i,j} \quad \{\text{M\$/TWe-h}\}$$

where, subscripts, i, j and X are indicated the reactor type, zone type and kinds of storage costs as above respectively.

E.5 Disposal Cost

Spent fuel shipping cost, disposal cost, HLW storage cost and HLW disposal cost are needed.

1. Spent Fuel Shipping Cost

$$Co_{SF,shp,i,j} = F_{SF,reposi,j} \cdot Co_{SF,shp,unit} \quad \{\text{M\$/y}\}$$

2. Total Spent Fuel Shipping Cost

$$Co_{SF,shp,tot} = \sum_{i,j} Co_{SF,shp,i,j} / C_{Rx,tot,deployed} \Big/ 8.76 \ \{\text{M\$/TWe-h}\}$$

3. Spent Fuel Disposal Cost

$$Co_{SF,disp,i,j} = C_{Rx,deploy,i} \cdot Co_{SF,disposal,unit} \cdot 8760 \quad \{\text{M\$/y}\}$$

4. Total Spent Fuel Disposal Cost

$$Co_{SF,disp,tot} = \sum_{i} Co_{SF,deployed,i} / C_{Rx,tot,deployed} \Big/ 8.76 \ \{\text{M\$/TWe-h}\}$$

5. High Level Storage Cost

$$Co_{HLW,sto} = M_{HLW} \cdot Co_{HLW,sto} \big/ C_{Rx,tot,deployed} \big/ 8.76 \ \{\text{M\$/TWe-h}\}$$

6. High Level Disposal Cost

$$Co_{HLW,disp} = M_{FP,reproc} \cdot Co_{HLW,disp} / C_{Rx,tot,deployed} / 8.76 \text{ \{M\$/TWe-h\}}$$

7. Total Disposal Cost

$$Co_{tot,disp} = Co_{SF,shp,tot} + Co_{SF,disp,tot} + Co_{HLW,sto} + Co_{HLW,disp} \text{ \{M\$/TWe-h\}}$$

E.6 Recycle Cost

1. Spent Fuel Shipping Cost

$$Co_{SF,ship,i} = Co_{SF,ship,unit,i} \cdot \left(C_{reproc,i,j} + C_{repproc,DUPIC,j}\right) / C_{Rx,deployed,tot} / 8.76$$

{M$/TWe-h}

$Co_{SF,reproc,unit,i}$=350 {$/kg}

2. Spent Fuel Reprocessing Cost

$$Co_{SF,reproc,i} = Co_{SF,reproc,unit,i} \cdot C_{reproc,i,j} / C_{Rx,deployed,tot} / 8.76 \text{ \{M\$/TWe-h\}}$$

$Co_{SF,reproc,unit,i}$=800 {$/kg}

3. Total Recycle Cost

$$Co_{SF,recyc,tot} = \sum_i Co_{SF,ship,i} + C_{SF,reproc,i}$$

E.7 MSR Cost

1. MSR Processing Cost

$$Co_{pro-fuel,MSR} = \left[\begin{matrix} C_{Rx,deployed,MSR} \cdot 1.06E6 \cdot DT \cdot 365 \cdot 24 \cdot \\ Co_{pro-fuel,unit,MSR} / 1.0E9 \end{matrix} \middle/ \left(C_{Rx,tot,deployed} \cdot DT \cdot 8.76 \right) \right],$$

{M$/TWe-h}

$Co_{pro-fuel,unit}$ =0.3 {mills/kWh(e)}

2. MSR Salt Inventory Cost

$$Co_{salt,inven,MSR} = \left[\begin{matrix} C_{Rx,deployed,MSR} \cdot DT \cdot 365 \cdot 24 \cdot \\ Co_{salt,inven,unit,MSR} / 1.0E9 \end{matrix} \middle/ \left(C_{Rx,tot,deployed} \cdot DT \cdot 8.76 \right) \right],$$

{M$/TWe-h}

$Co_{salt,unit}$ =0.04{mills/kWh(e)}

3. MSR Salt Makeup Cost

$$Co_{salt,makeup,MSR} = \left[\begin{matrix} C_{Rx,deployed,MSR} \cdot DT \cdot 365 \cdot 24 \cdot \\ Co_{salt,makeup,unit,MSR} / 1.0E9 \end{matrix} \middle/ \left(C_{Rx,tot,deployed} \cdot DT \cdot 8.76 \right) \right],$$

{M$/TWe-h}

$Co_{salt,makeup,unit}$ =0.05 {mills/kWh(e)}

4. MSR Moderator Cost

$$Co_{moderator,MSR} = \left[\begin{matrix} C_{Rx,deployed,MSR} \cdot 1.0E6 \cdot 365 \cdot 24 \\ Co_{moderator,unit,MSR} / 1.0E9 \end{matrix} \middle/ \left(C_{Rx,tot,deployed} \cdot DT \cdot 8.76 \right) \right],$$

{M$/TWe-h}

$Co_{moderator,unit}$ = 0.1 {mills/kWh(e)}

5. Total Cost of MSR Operation

$$Co_{op,MSR} = Co_{moderator,MSR} + Co_{salt,inven,MSR} + Co_{salt,makeup,MSR} + Co_{moderator,\,fuel-rpoc,MSR}$$

III. CONCLUSION

Recently, DYMOND code has been modified to apply DUPIC, MSR and RTF fuel cycle analysis. It can predict the cost for material flow path as well as mass flow. In this report, all the models in the modified DYMOND code were described in detail. Also, three kinds of different fuel cycles such as DUPIC, MSR and RTF were reviewed and predicted the materials and the costs in each path of mass flow for next 100-year Global Energy Park. The results attached as an appendix.

The present modeling report was prepared in order to give users all the information which were modeled in modified DYMOND code. But, DYMOND code will be continued developing for another application such as ACR, SCWR, ADS or new innovative nuclear systems.

APPENDIX I : Results of DUPIC Scenario

DUPIC Scenario

1. Scenario Description and DUPIC Attributes

The DUPIC (Direct Use of Spent PWR Fuel in CANDU Reactors) scenario involves partial recycle of LWR spent fuel (SF) materials in CANDU reactors. The LWR SF is mechanically separated into two major streams: (1) the UO_2 with fission products and actinides and (2) the SF cladding. The UO_2 with fission products and actinides is fabricated into CANDU fuel assemblies. These fuel assemblies are used to fuel CANDU reactors and then disposed of as SF in a geological repository.

The scenario assumes that between the years 2010 and 2100, CANDU reactors are built at a sufficient rate (for example, for achieving the ratio of 2 PWRs and 1 CANDU reactor) so that almost all LWR SF including existing PWR and BWR spent fuels can be recycled into CANDU reactors from the year 2020. The rate of building new CANDUs is so that all LWR SF is recycled

DUPIC reactors will be built at a sufficient rate that the SF from two LWRs will be used to fuel one DUPIC reactor. However, to reach a balance between SF generated from LWRs and recycled SF going to DUPICs, the initial LWR SF (250 kt) and SF generated from existing LWRs will need to be taken into account. Thus, to reach this balance, an initial high build up of DUPICs is assumed followed by gradual decrease in buildup rate until a steady rate of buildup is achieved. The corresponding rates that are used in this scenario are as follows. Between the years 2015 and 2020, DUPICs are ordered at rate which will satisfy 33% of the new and replacement electrical capacity, followed by a rate of 30%, between 2020 and 2025. The share decreases to 25% between 2025 and 2037, and beyond 2025, a constant share of 22% is assumed. The 22% rate is based on an LWR SF discharge rate that is ½ of the DUPIC discharge rate. Based on Tables A.1, which shows the attributes for DUPIC type reactors, the SF discharge rate is about 46000 kg/y. This is double the LWR discharge rate used in the scenario (about 23000 kg/y). Based on reactor powers of 1.3 GWe and 0.713 GWe, for LWRs and DUPICs, respectively, the pace of building two LWRs for each DUPIC will satisfy about 22% of the new and replacement electrical capacity.

To simplify the scenario simulation, a number of assumptions were made as follows.

- The existing CANDU capacity of about 5% (about 18 GWe) of the total capacity, before the year 2020, is assumed to be part of the existing LWR capacity. Using the DUPIC fuel in the existing CANDUs will have small effect in depleting the existing SF (about 250 kt) and the newly generated SF from new and old LWRs. Thus, no special modeling for the CANDU reactors before the year 2020 is included in the DYMOND code.
- After 2020, the DUPIC fuel fabricated from LWR SF is only used in new DUPIC type reactors that are built after the year 2020.
- No LWR SF will be sent to repository since it will be needed for feeding DUPIC type reactors.

- To be consistent with other scenarios, SF storage charge is a one time charge of $50 per kg HM + $5 per kg-year of storage (this cost has been used for all types of reactors in other scenarios) instead of using $32 per kg-year given in Table A.3. By the year 2100, the use of $32 per kg-year would have increased the total cost by about 10%.

2. <u>Discussion</u>

A summary for the results of this scenario is shown in Figures 1-3. Figure 1 shows the known and unknown U resources consumption for this scenario. Those resources will almost be depleted by the year 2050 (it will be completely exhausted by the year 2053). The continuing buildup of LWRs in addition to the existing LWRs will be enough to consume those resources. By the year 2100, about 33,000 kt of new U discoveries will be needed. This amount is about 20% less than the new resources needed during the once-thru LWR scenario.

The SF from this scenario includes SF from the LWRs (all in storage) and SF from the DUPIC reactors (in storage or repository). Figure 1 shows those different amounts of SF. The UO_2 SF from the LWR will remain the major part of the total SF as it reaches a peak around the year 2040. Beyond 2040, the high buildup of new DUPICs starts to consume the LWR SF at a rate that is larger than its production rate, and the pile of LWR SF starts to decrease. This is the case, although the rate of DUPICs buildup will be decreasing (if the rate of build up is kept at the earlier higher levels, there will not be enough LWR SF to feed the existing DUPICs). Gradually, the LWR SF contribution to the total SF decreases until the year 2052. Beyond that, the SF from the DUPIC reactors will be the major source of SF. Notice that by the year 2085, all excess LWR SF will be consumed, and LWR SF will be generated at a rate that is similar to the rate of its recycling in DUPIC reactors. In this case, the amount of LWR SF accumulating in storage will almost be permanently eliminated. By the year 2100, the total amount of SF from LWRs and DUPICs will be about 2500 kt. This amount is about 40% less than the 4100 kt of SF associated with the once-thru LWR scenario.

As expected, the enrichment rate for this scenario is lower than the rate for the once-thru LWR scenario as a result of the reduction on demand for the enriched U that is used in LWRs. By the year 2100, about 20% reduction in enrichment rate is expected. The fabrication rate, however, increases for this scenario compared to the LWRs-only scenario. DUPIC reactors consume double the amount of fuel that is consumed by LWRs, which requires more fuel fabrication for this type of reactors. About 40% increase in the total fabrication rate is expected by the year 2100. The capacity for fabricating the DUPIC fuel is also shown in the figure. This capacity is initially determined by forecasting the DUPIC fuel requirement in the future (within the 5 years it take to build a fabrication plant) based on the history of the stored LWR SF within the previous 5 years. Beyond the year 2025, this forecast is based on the DUPIC fuel requirements in the previous 5-10 years. Based on this rate of building fabrication plants (each has 0.4 kt/y capacity), just enough fuel will be fabricated from the LWR SF to meet the DUPIC reactors requirements.

Figure 2 also shows the electricity production market share of the DUPIC reactors compared to the LWRs. By the year 2100, this share will represent about 18% of the total electricity generation capacity.

Cost estimates for this scenario are shown in Figure 3. The mining and enrichment costs are the major cost factor for this scenario similar to the once-thru LWR scenario. A 20% reduction is expected by the year 2100 compared to the LWR base scenario. The cost of fabrication will increase compared to the LWRs-only scenario, because of the larger amounts of fuel to be fabricated for the DUPIC reactors (this cost will be more than double by the year 2100). Additional costs for this scenario, over the base scenario, are the recycling costs (mainly include the cost of shipping the LWR SF to the fabrication plants), and the cost of disposing of the HLW from the fabrication plants (the 0.5% lose of fuel during fabrication). Compared to the base scenario, the increase in total cost due to those additional costs and the increase in fabrication costs will almost offset the decrease in the enrichment and mining costs. This leads to total cost estimates that are similar to the cost estimates for the base LWR scenario.

The differences between the performance of this scenario and the base LWR scenario are also shown in Figure 3, through the normalized indexes. As discussed before, the cost of this scenario ends up about the same as the cost for the base scenario, which is shown by a cost index value that is about one. The ore index reaches about 0.82 by the end of simulation, reflecting the savings made in U resources through using the DUPIC reactors (about 18% savings). Another benefit from this scenario is the realized reduction in the total amounts of SF. This is shown through the SF index, which will go down to about 0.6 by the year 2100. Thus, about 40% reduction in the existing amounts of SF by 2100, compared to the base scenario, will be achieved as a result of the LWR SF recycling in the DUPIC reactors. Finally, this reduction in waste will lead to reductions in the amounts of Pu and MA in the SF, which are reflected in the Pu and MA indexes. The Pu index is reduced to about 0.5. This is a result of both the 40% reduction in waste and the lower Pu fraction in DUPIC SF (~ 1%) compared to the Pu fraction in LWR SF (~ 1.2%). The MA index is reduced to about 0.64 as a result of the reduction in SF and the slight increase in MA fraction in DUPIC SF compared to LWR SF (~ 0.163% compared to ~ 0.149%, respectively). In general, it seems that this scenario will lead to significant reduction in both the required U resources and the amount of generated SF by the year 2100.

Appendix A:

DUPIC Attribute Sets

Table A.1 Power Plant Attribute Values for the DUPIC Fuel Cycle

(B) Attribute List for DUPIC Fuel in CANDU Plant(CANDU DUPIC Fuel)

Power Plant (In case that the DUPIC fuels are used in existing CANDU reactor)

- Rating (MWe) : 713
- Station efficiency (η) : 33%
- Construction lead time (y); 4
- Lifetime (y) : 40
- Capacity factor : 90%

Working Inventories:

- $\frac{\text{kg IHM}}{\text{MW}_{th}}$: 86,640/2,159 = 40.13
- Mass loading fraction : DUPIC fuels are fully loaded in whole CANDU core.
- Average Enrichment : 0.601% for Pu239, 0.098% for Pu241, 0.574% for U235
  $\rightarrow$ total : 1.273 for DUPIC fuel.
  * DUPIC fuels are assumed to be directly fabricated with PWR spent fuels with 50,000 MWD/MTU of burnup.
- Fertile material (U) : U238

Mass Flows:

- Refueling Interval : on-line refueling
- # of reload batches : NA
- Ave discharge burnup $\frac{\text{MW}_{th}\ \text{days}}{\text{kgIHM}}$: 15,400
- Mass fractions in Discharge Fuel :

  $\frac{\text{kg Pu}}{\text{kgIHM}}$, = 0.987%, $\frac{\text{kg TRU}}{\text{kgIHM}}$, = 1.15%

  $\frac{\text{kg U235}}{\text{kgIHM}}$, = 0.15%, $\frac{\text{kg U233}}{\text{kgIHM}}$ =0

Lag time;

- Lag from discharge to processing of spent PWR fuel : 10 years
- Lag from processing to refab. : 0.5 year
- Lag from refab. to reload : 0.5 year

Table A.2 DUPIC Fuel Fabrication Facility Attribute Set

	DUPIC Fuel Fabrication facility
Construction Lead Time, y	5
Licensing Lead Time, y	2
Plant Lifetime, y	40
Thruput Tonnes HM/y Rate, y	400
Storage Prior to Processing, y	10
In Process Dwell Time, y	½
Total out of Reactor Time, y	1½
Loss to Waste per Recycle/Refab pass (%)	0.5

Table A.3 Unit Costs for Fuel Cycle Components Relating to CANDU-DUPIC Fuel†

Component	**Description**	**Unit cost**			**Unit**
		Lower bound	*Nominal Value*	*Upper bound*	
$Cost_{DUPICfab}$	DUPIC fuel fabrication	448	616	784	$/kgHM
$Cost_{DUPICinstore}$	DUPIC spent fuel interim storage[1]	21	32	42	$/kgHM.year
$Cost_{DUPICsftransport}$	DUPIC spent fuel transport[2]	22	28	33	$/kgHM
$Cost_{DUPICgeo}$	DUPIC spent fuel conditioning and disposal[3]	73	167	279	$/kgHM

† All costs are expressed in 2000-dollars. All unit costs related to DUPIC fuel cycle are well described in the reference paper below. In the paper, unit costs of PWR fuel cycle are also described, but the values for transportation, interim storage, and disposal are a little different from the values described in the Table A.3-1 (Unit Costs for Fuel Cycle Components Relating to LWR-Reactor) of Crosscut Group Report. In order to maintain the consistency of the DUPIC data with PWR data shown in the report, those data are converted by the ratios of DUPIC values to PWR values considering original Table A.3-1's values.

1 In order to be consistent with the other scenarios, we have used the base LWR spent fuel storage cost instead of the reference value used in the table.

2 DUPIC spent fuel transport cost is estimated to be 28 $/kgHM, which is about 55% of LWR spent fuel transport cost in the paper.

[3] DUPIC spent fuel conditioning and disposal cost is estimated to be 167 \$/kgHM, which is about 56% of LWR spent fuel conditioning and disposal cost in the paper.

Reference

Ko, W.I., Choi, H.B. and Yang, M.S., “Economic Analysis on Direct of Spent Pressurized Water Reactor Fuel in CANDU Reactors (IV) – DUPIC Fuel Cycle Cost”, Nuclear Technology, Vol. 134, May 2001

Figure 1.

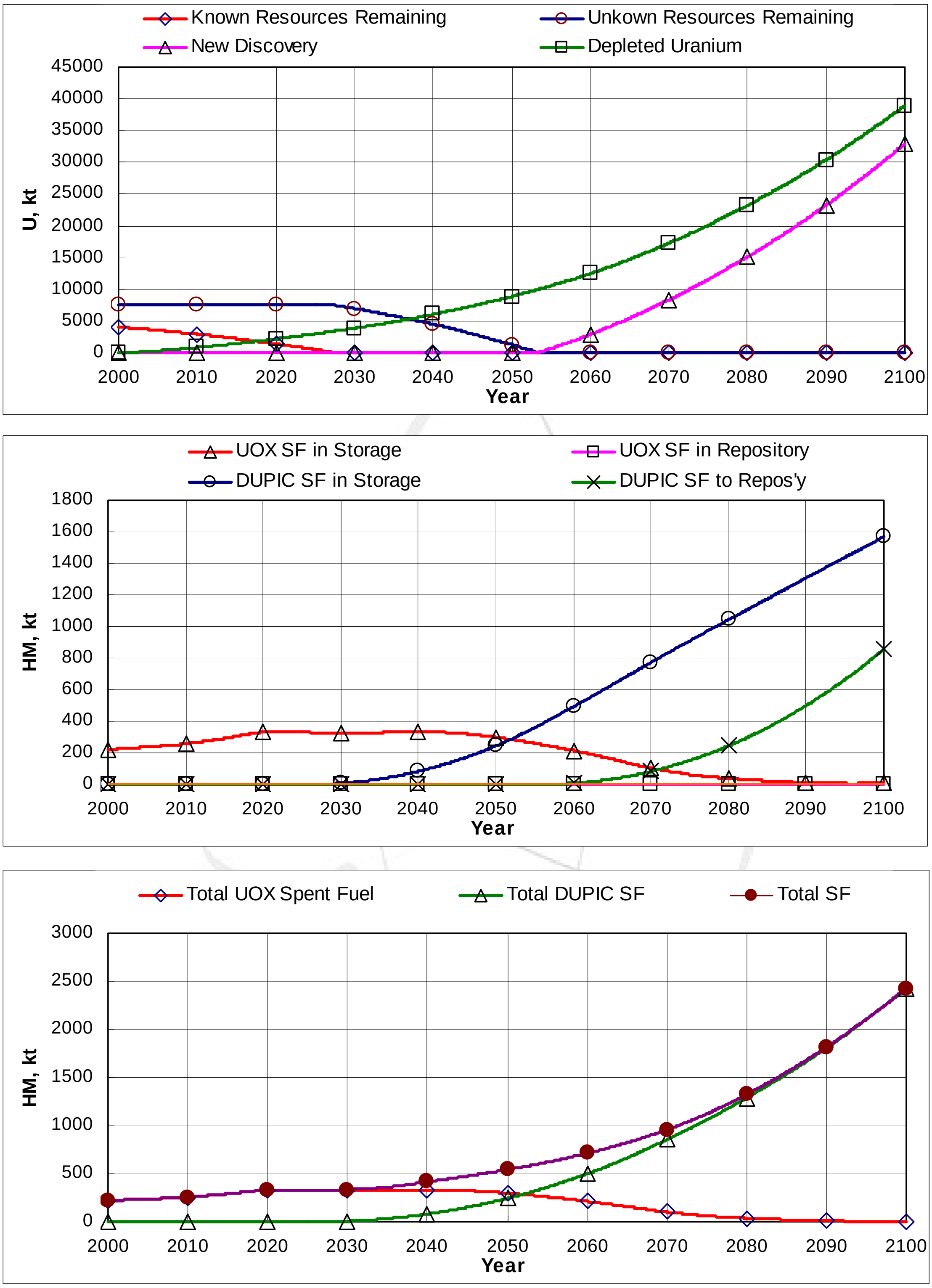
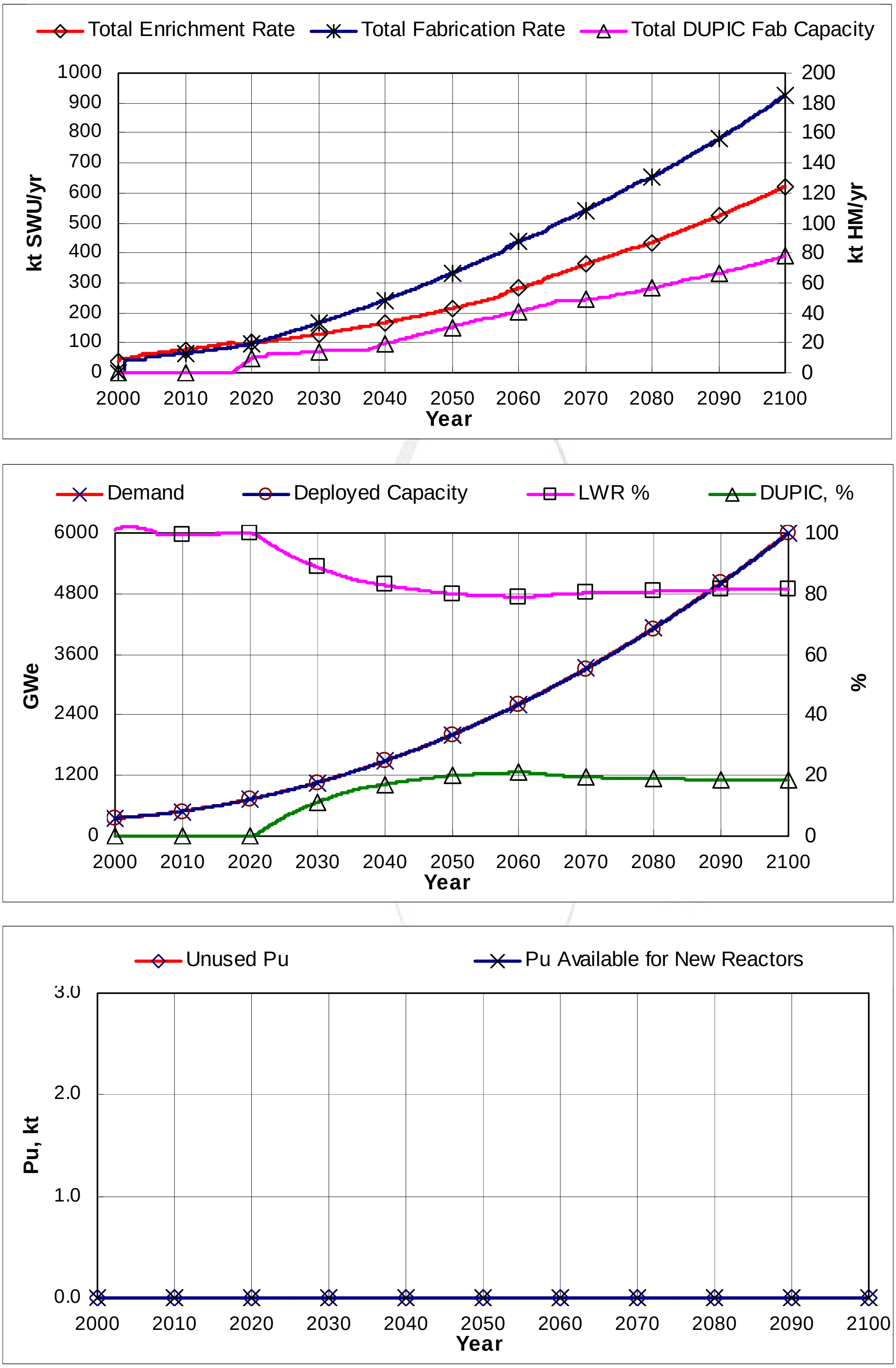
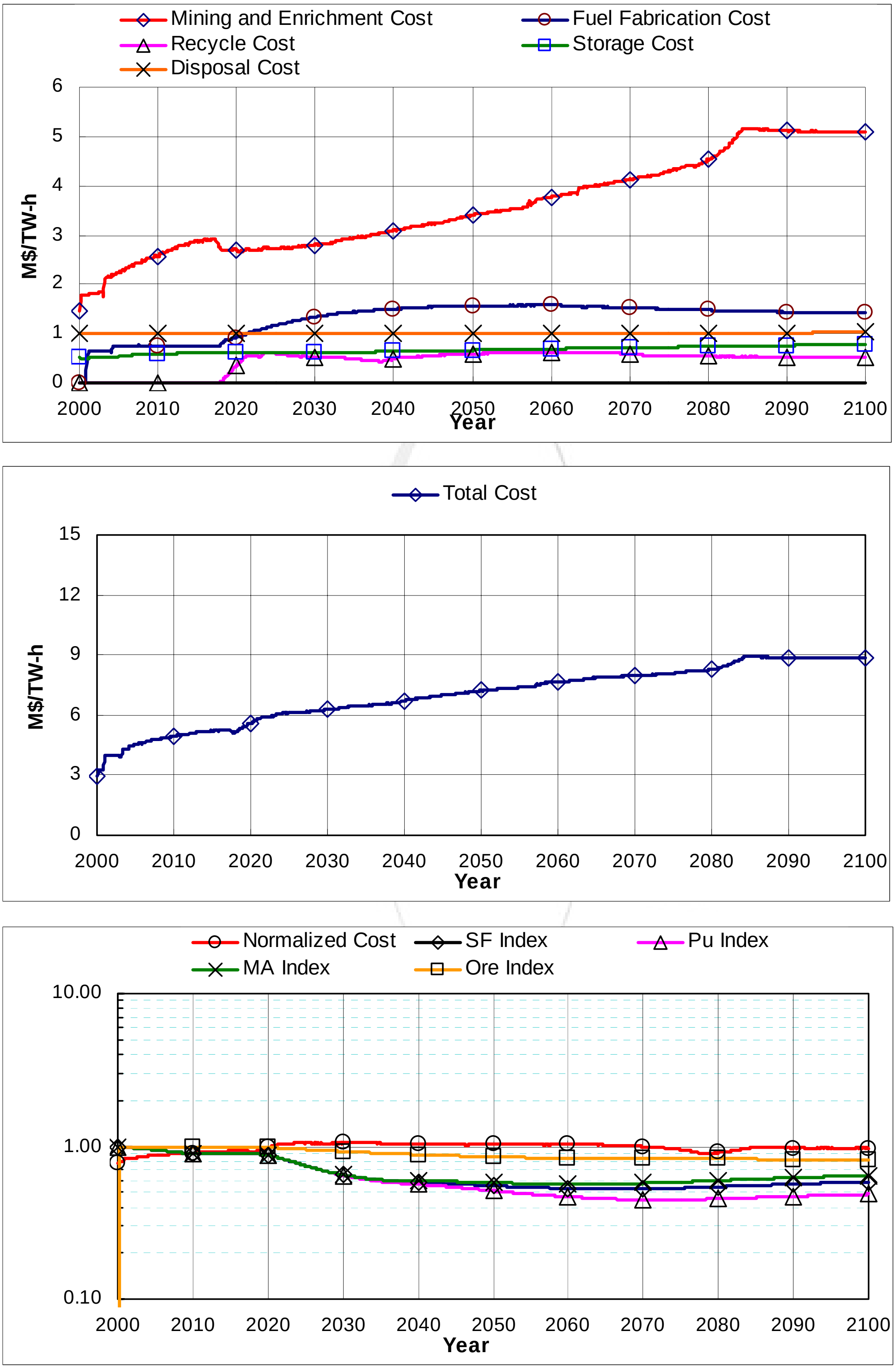

Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
45000
40000
35000
30000
25000
20000
15000
10000
5000
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
UOX SF in Storage
UOX SF in Repository
DUPIC SF in Storage
DUPIC SF to Repos'y
HM, kt
1800
1600
1400
1200
1000
800
600
400
200
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
Total UOX Spent Fuel
Total DUPIC SF
Total SF
HM, kt
3000
2500
2000
1500
1000
500
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

Figure 2.



Total Enrichment Rate
Total Fabrication Rate
Total DUPIC Fab Capacity
kt SWU/yr
kt HM/yr
1000
900
800
700
600
500
400
300
200
100
0
200
180
160
140
120
100
80
60
40
20
0
2000 2010 2020 2030 2040 2050 2060 2070 2080 2090 2100
Year
Demand
Deployed Capacity
LWR %
DUPIC, %
GWe
%
6000
4800
3600
2400
1200
0
100
80
60
40
20
0
2000 2010 2020 2030 2040 2050 2060 2070 2080 2090 2100
Year
Unused Pu
Pu Available for New Reactors
Pu, kt
3.0
2.0
1.0
0.0
2000 2010 2020 2030 2040 2050 2060 2070 2080 2090 2100
Year

Figure 3.



Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
M$/TW-h
0
1
2
3
4
5
6
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
Total Cost
M$/TW-h
0
3
6
9
12
15
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
0.10
1.00
10.00
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

APPENDIX II : Results of MSR Scenario

Molten Salt Reactor Scenario

1. <u>**Scenario Description and MSR Attributes**</u>

In this scenario, MSR is introduced in 2030, where 5% of the new plants (replacement and growth) are MSRs. In 2031, 10% of the new plants are MSRs. This linear market-fraction penetration extends until all reactors are MSRs. The MSR is assumed to have 44% efficiency and a capacity factor of 90%. All MSRs starting up contain an initial inventory of 127,460 kg of thorium per 1 GW(e) capacity. The fissile material for startup of new MSRs from 2030 to 2050 is 19.9 wt % enriched uranium. The startup inventory is 3115 kg of ^{235}U in 15,653 kg of total U for 1 GW(e) capacity. After 2050, all new 1-GW(e)-capacity MSR plants that startup are loaded with (1) 127,460 kg of thorium, (2) 3115 kg of recycle LWR plutonium plus associated higher actinides (i.e., Pu and minor actinides or Pu+MA) and (3) 15,653 kg of total U. In a MSR, most of the plutonium is rapidly burnt out and replaced with ^{233}U. The depleted uranium is added with the startup plutonium to denature the ^{233}U as it grows in. There is no enriched uranium for MSRs started on plutonium. Plutonium adds safeguards and technical complications to startup of a MSR; thus, the use of plutonium for startup is delayed for 20 years while experience is gained with MSRs.

At the end of plant life, each retired MSR is replaced with a new MSR that uses the fuel load from the decommissioned MSR. MSRs used to replace retiring MSRs do not get added enriched uranium or recycle plutonium. In practice, the fuel salt with fuel (fissile and fertile) is recycled into replacement reactors to (1) recycle the fuel, thorium, and the expensive ^{7}Li and (2) avoid disposal of the old salt with its beryllium.

For each GW(e)-year of electricity generated, the reactors require an added feed of 801 kg of thorium and 155 kg of depleted uranium. This is independent of what the startup fissile material is (enriched uranium or plutonium). Within a few years of startup, the fuel composition in the salt is essentially independent of what fissile material was used to start the reactor.

In performing the scenario simulation, a number of assumptions were made as follows.

- The composition of the spent fuel (SF) at the end of the reactor lifetime is assumed to be the same as the composition used at the reactor startup except for the fissile material content. The fissile material used to startup the reactor (U235 or Pu+MA) is replaced with U233 as the reactor reaches equilibrium.
- The processing of the molten salt during operations results a processing loss of about 526 grams of Pu and MA, which will go to the high level waste (HLW) stream.
- The fission products (FP) are removed from the reactor during operations at the same rate that Th and depleted uranium (DU) are fed to the reactor. That is, the rate of FP removal is 801 kg + 155 kg – 0.526 kg = 955.4 kg/year.

- LWR fuel reprocessing is assumed to start 2045 and the reprocessing plants are built at a rate, which will meet the demand of MSRs for Pu and higher actinides beyond the year 2050.
- The composition of Pu+MA startup fuel is assumed to contain proportions of Pu to MA that are similar to the proportions found in the LWR spent fuel.
- APUREX process is used to reprocess the LWR SF.
- A cost of 95$/kg Th is assumed here. This cost is based on average of $82.5/kg for 99.9% purity and $107.25 for 99.99% purity reported in reference 1. The Th conversion cost is assumed to be the same as the U conversion cost. Other costs, such as the storage cost, and HLW storage and shipping cost are assumed to be the same as the costs used in all scenarios.

2. <u>Discussion</u>

A summary of the results for this scenario is shown in Figures 1-3. Figure 1 shows a depletion of the known and unknown uranium resources by the year 2050. This is similar to the once-thru LWR scenario, since the LWRs remain dominant in power production until this point in time. However, by the year 2100 a large reduction in the required new uranium resources is achieved (about 10,000 kt are needed compared to about 42,000 kt for once-thru LWR scenario).

As part of the MSR scenario, no SF is sent to the repository, as it remains in storage until it is sent to the reprocessing plants. The scenario shows a continuing increase in the stored LWR spent fuel, but at a much slower rate than that associated with the once-thru LWR scenario. By the year 2065, the amount of SF in repository and storage associated with the once-thru LWR scenario is about 1575 kt compared to a stored 1145 kt for the MSR scenario. Beyond 2080, the amount of LWR SF associated with the MSR scenario starts to decrease as more SF is reprocessed to meet increased buildup of MSRs. By the year 2100, there are about 1000 kt of LWR SF remaining in storage compared to about 4000 kt of LWR SF in storage and repository in the case of once-thru LWR scenario. However, as a by-product of the reprocessing associated with the MSR scenario, there are additional 1200 kt of burned uranium which will be generated.

Figure 2 shows decrease in the rates of fuel fabrication and uranium enrichment as the MSRs are introduced. The enrichment rate decreases since enriched uranium is used in the MSRs between the years 2030 and 2050 to only startup it up and it is not used as a feed during the reactors operations. Beyond 2050 no enriched uranium is needed to start the MSRs. By the year 2100 only about 25 kt SWU/yr enrichment rate is needed compared to a rate of about 750 kt SWU/yr in the case of once-thru LWR scenario. The fabrication rate also decreases after the MSRs introduction since fuel fabrication is needed for the MSRs. Only fabrication of LWR fuel is needed, which decreases to a rate of about 5 kt HM/yr compared to about 130 kt HM/yr in the case of once-thru LWR scenario. The figure also shows the reprocessing rates of the LWR SF. The reprocessing plants are built at a rate that assures the presence of enough Pu supplies to meet the growing buildup of MSRs. The Pu that is available for use in MSRs is shown in the figure. Notice that the appropriate increase in building up the reprocessing capacity kept the amounts of available Pu from being too excessive or being short of meeting the MSR needs. Finally, this figure shows the contribution of the MSR to the total electricity production capacity. The MSRs reach a 50-50 market share by the year 2067 and achieves about 95% market share by the year 2100.

Cost estimates for the MSR scenario are shown in Figure 3. The mining and enrichment cost remains the major cost factor for this scenario as it was for the once-thru LWR scenario. The amount of mined Th ore is much smaller than the amount of mined uranium ore. This is the case, even by the year 2100 where the majority of the electricity production is attributed to MSRs. This is a result of the low amounts of Th needed over the life of a MSR compared to the amounts of U needed over the life of a LWR (about 175 tons of Th compared to about 1400 tons of U over a 60 years lifetime of a 1GWe MSR and 1.2 GWe LWR, respectively). Enrichment and conversion costs are usually minor compared to the mining costs. Fabrication and storage costs are costs associated with the operating LWR, since only LWR fresh fuel and LWR SF need to be fabricated and stored, respectively. Both costs decrease with time after the introduction of the MSRs. Beyond 2050, after the introduction of the Pu based MSRs, the recycling cost (based on APUREX process for Pu and MA extraction from LWR SF) becomes a major cost factor. This is especially after the year 2080, as it exceeds the mining costs. For this scenario, the disposal cost corresponds to the cost of temporary storing and transporting the HLW associated with the LWR SF reprocessing, and the repository cost. A cost of 1 mills/kWh is still charged for permanent storage of this HLW in the repository. The costs of temporary storage and transportation of the HLW are very small compared to the repository cost. Finally, a cost that is unique to the MSR is considered here. This cost is related to the salt processing in order to remove the FP and recycle the salt back into the reactor. This cost might be considered as part of the capital cost of the reactor or its operations cost. However, it is considered here as a cost of the fuel cycle in order to see its effect on the total fuel cycle cost. It is assumed here that this cost is 1 mill/kWh. A recent study of the cost of the denatured MSR(2) did not include the processing cost, since the rector operations in this case does not require the FP removal from the salt. A 1970 estimate of this cost for the MSBR(3) is about 0.3 mills/kWh and adjusting it for the year 2000 cost, we assumed it to be 1 mills/kWh. In this case, as shown in Figure 3, the processing cost increases with time as more MSRs are introduced. Beyond the year 2090, this cost exceeds the costs attributed to the other parts of the fuel cycle. The total costs associated with the MSR scenario are also shown in the figure. This cost is smaller than the cost associated with the once-thru LWR scenario as discussed next in relation to the normalized indexes.

Figure 3 shows the normalized indexes, which compares the MSR scenario to the once-thru LWR scenario. As mentioned before, the cost of the MSR scenario is less than that of the once-thru LWR, mainly as a result of decrease in the mining cost. The normalized cost index goes down gradually with the introduction of the MSRs until it reaches about 0.35 by the year 2100. Notice that starting the year 2000, the cost index is slightly less than one since no SF is shipped to the repository in this scenario, which eliminates the SF shipping cost. The SF will be used to fuel the MSRs, thus, the SF index is zero. The ore index also decreases with time after the MSR introduction because of the decreased demand for the U ore as discussed before. By the year 2100, the ore index will be reduced to about 0.4. The Pu and MA indexes increase above zero as the LWR SF processing and the MSRs operations start. This is a result of the losses during the LWR SF reprocessing operations and the .losses during the recycling of the MSR salt. Finally, there are no noticeable changes in the indexes as the early built MSRs go off line (starting the year 2095). This is a result of the assumption that core of the a MSR at the end of life is used to operate a new MSR. Ignoring the lag in time used to build the new reactor, this will be equivalent to a reactor that is running indefinitely.

References:

1. Chapter 1, “Introduction: Conceptual Framework & Issues,” Gen-4 Fuel Cycle Crosscut Group, November 1, 2001, section 1.3.2.
2. R. W. Moir, “Cost of Electricity from Molten Salt Reactors,” to appear in Nucl. Tech., Vol. 138, April 2002.
3. A. M. Perry and H. F. Bauman, “Reactor Physics and Fuel-Cycle Analysis,” Nucl. Appl. Tech., 8, 208 (1970).

<u>Appendix B:</u>
Nuclear Power Plant Attribute Sets

Table A.1 Power Plant Attribute Values for the MSR Fuel Cycle

<u>Attribute List for MSR Fuel</u>

Power Plant
- Rating (MWe) : 1000
- Station efficiency (η): 44%
- Construction lead time (y); 4
- Lifetime (y) : MSR is assumed to continue beyond the usual 60 years lifetime which is equivalent to reusing the molten salt with its heavy metal in a new reactor
- Capacity factor : 90%

Working Inventories:

- $\frac{\text{kg IHM}}{\text{MW}_{th}}$: 143,113/2273 = 63
- Mass loading fraction: MSR fuel is fully loaded in whole MSR core.
- Average Enrichment : 19.9% U235/(Total U) for startup with U235
  19.9% (Pu+MA)/(U238+Pu+MA) for startup with Pu + MA
  (i.e., 3115/143,113 = 2.2% enrichment, for both cases)
- Fertile material (U) : Th, U238

Mass Flows:
- Refueling Interval : on-line refueling
- # of reload batches : NA
- Ave discharge burnup: fuel remain in the core during the reactor lifetime
- HM feed during operations: 801 kg Th/year, 155 kg DU/year
- Processing waste: 526 gm Pu+MA (all goes to HLW and end up in repository with FP)
- FP removed from reactor: 801 + 155 = 956 kg/year
- Mass fractions in Discharge Fuel : Assumed to be the same as the startup fractions (the startup fractions used here are actually the equilibrium fractions and all generated actinides are consumed during operations and only a small fraction of the Pu and MA are lost during the processing of the molten salt)

Figure 1.

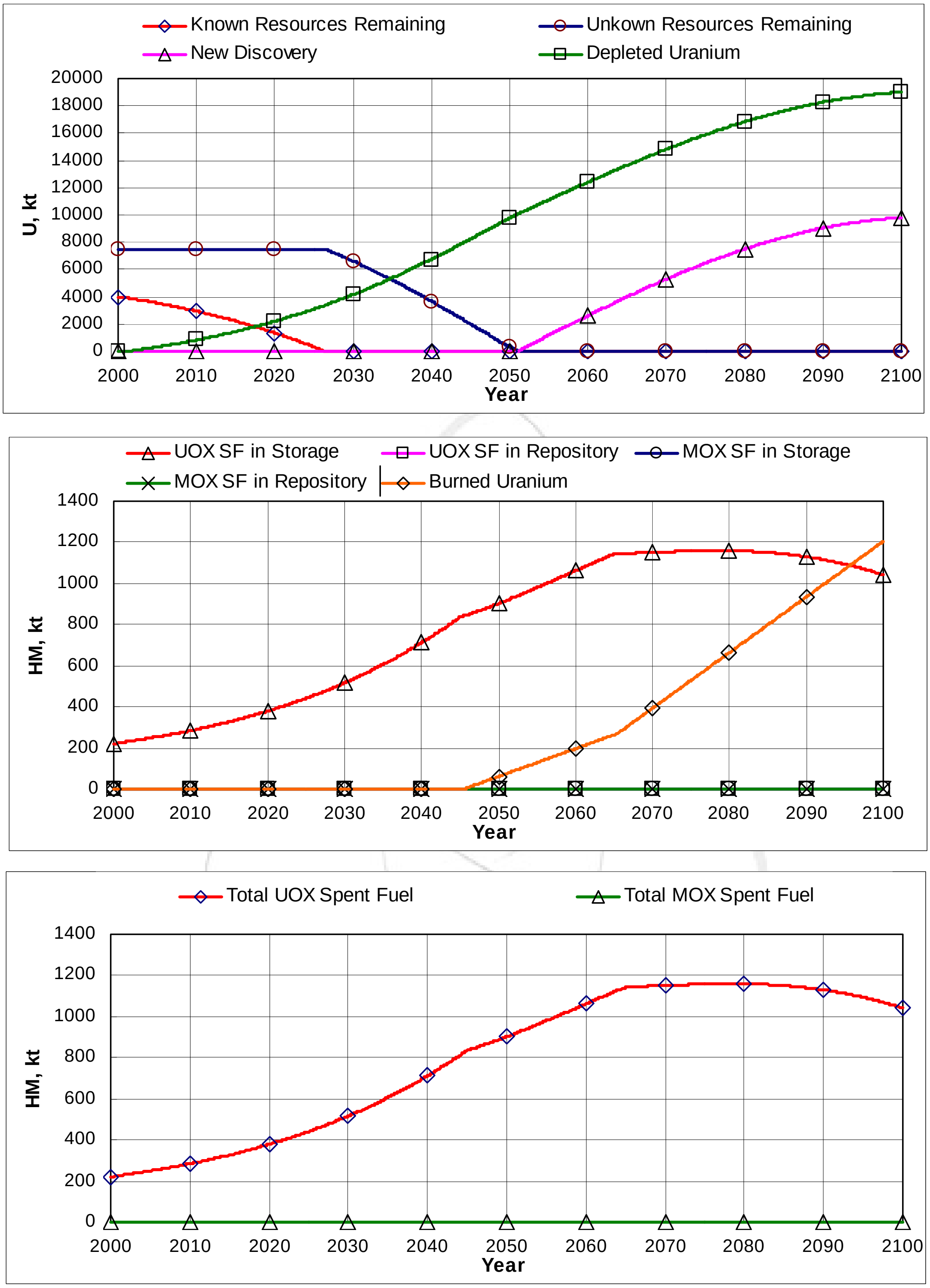
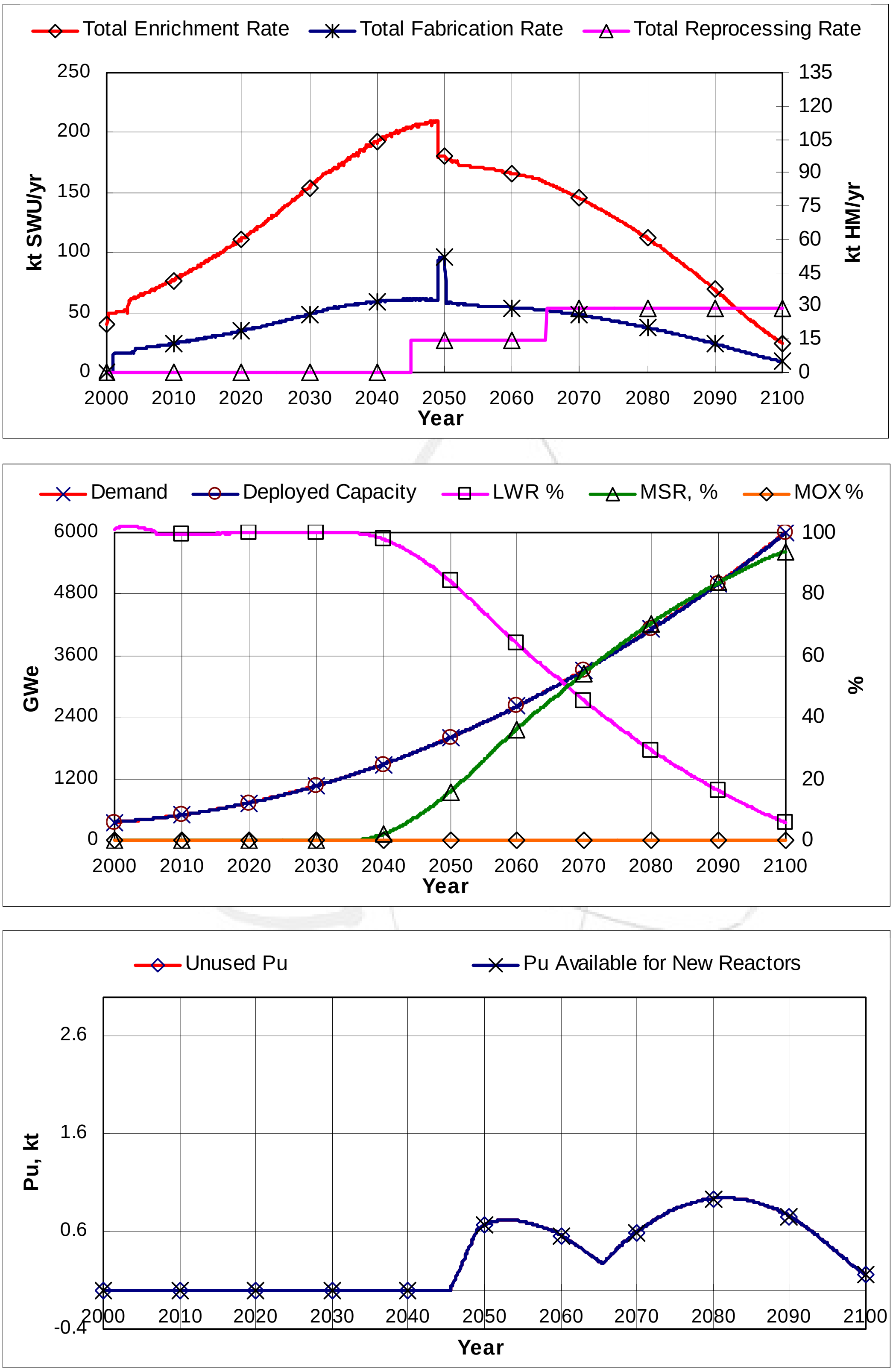
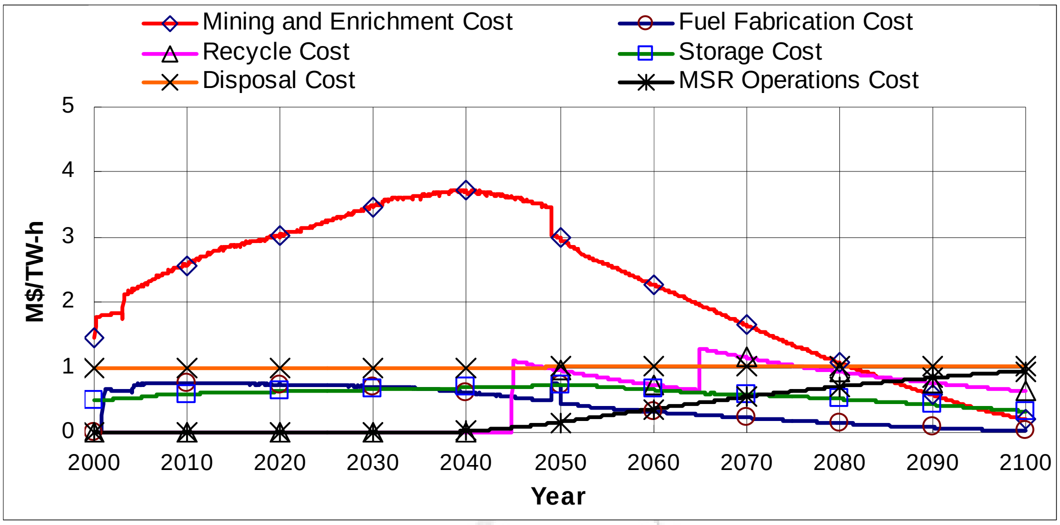

Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
20000
18000
16000
14000
12000
10000
8000
6000
4000
2000
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
UOX SF in Storage
UOX SF in Repository
MOX SF in Storage
MOX SF in Repository
Burned Uranium
HM, kt
1400
1200
1000
800
600
400
200
0
Year
Total UOX Spent Fuel
Total MOX Spent Fuel
HM, kt
Year

Figure 2.

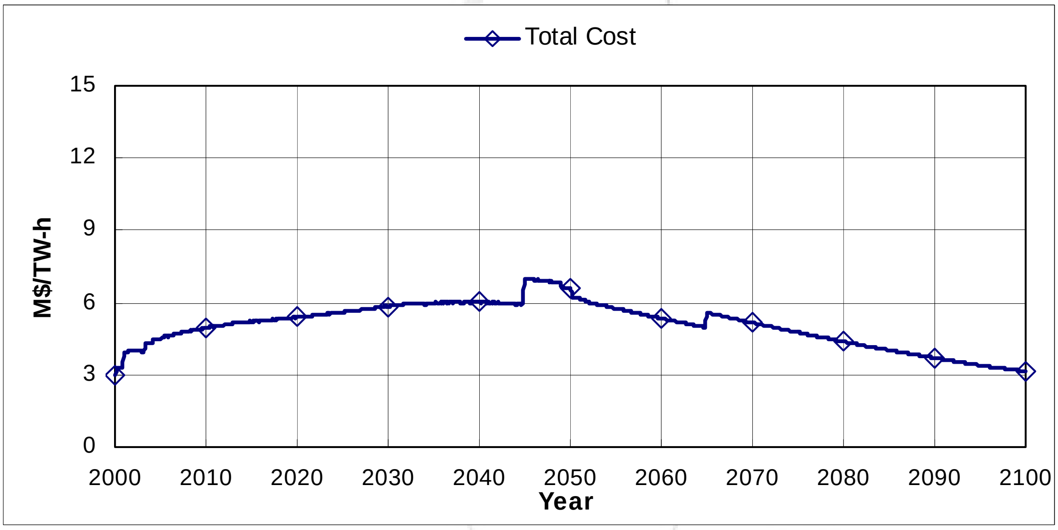


Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
250
200
150
100
50
0
kt SWU/yr
135
120
105
90
75
60
45
30
15
0
kt HM/yr
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
Demand
Deployed Capacity
LWR %
MSR, %
MOX %
6000
4800
3600
2400
1200
0
GWe
100
80
60
40
20
0
%
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
Unused Pu
Pu Available for New Reactors
2.6
1.6
0.6
-0.4
Pu, kt
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

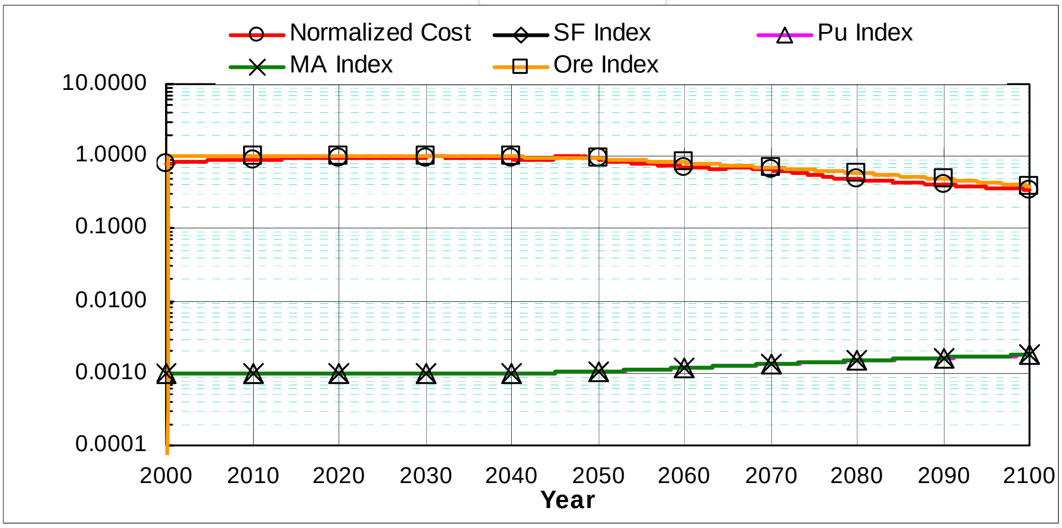
Figure 3.



Mining and Enrichment Cost
Fuel Fabrication Cost
Recycle Cost
Storage Cost
Disposal Cost
MSR Operations Cost
M$/TW-h
0
1
2
3
4
5
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year



Total Cost
M$/TW-h
0
3
6
9
12
15
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year



Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
10.0000
1.0000
0.1000
0.0100
0.0010
0.0001
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

APPENDIX III : Results of RTF Scenario

Once-through LWR Thorium Fuel Cycle Scenario

1. <u>Scenario Description and RTF Attributes</u>

This scenario calls for starting to adopt the Radkowsky thorium fuel cycle (RTF) by the year 2020 and to fully adopt it by 2040. To implement the scenario, it was assumed that starting the year 2020, 20% of the new plants (replacement and growth) will be RTF reactors, 40% share by 2025, 60% by 2030, 80% by 2035, and 100% by 2040. The mass flow data for the reactor are based on data in references 1-4. The RTF type subassemblies consist of two regions; internal seed region and outer blanket region. The seed region contains U-Zr (~20% enriched U) metallic alloy and the blanket region contains ThO2UO2 (~ 10%UO2 at ~20%enrichment). The breeding ratio for the reactor is ~ 1, and the blanket part of the subassembly is removed from the reactor after ~ 10 years, while the seed part remains for 3 years. This leads to the higher blanket burnup shown in the reactor attributes table (Table 1) while the seed fuel reaches a burnup similar to the LWRs burnups. U233 is produced in the blanket fuel and it ends up in the repository as part of the spent fuel (SF).

Detailed cost data for the elements of this cycle are not available, and it was assumed that the time and cost data for the RTF type reactor are the same as those for an LWR. For example, the reactor construction time, licensing time, SF storage time, and SF cooling time are the same as those for the LWRs. Also, cost parameters such as the fabrication cost, the SF storage cost, and the disposal cost (including the SF shipping cost) are the same as the corresponding LWR costs. The Th cost is the only additional cost considered in this scenario. The cost is assumed to be about 95$/kg Th, based on average of $82.5/kg for 99.9% purity and $107.25 for 99.99% purity reported in Chapter 1, "Introduction: Conceptual Framework & Issues," Gen-4 Fuel Cycle Crosscut Group, November 1, 2001, section 1.3.2. The mining cost is assumed to be constant here, although the reference suggests that the mining cost can actually decrease if levels of consumption will increase.

2. <u>Discussion</u>

The DYMOND code tracked the seed and blanket fuel separately since its mass flows are independent from each other. A summary of the results for this scenario is shown in Figures 1-3. Figure 1 shows a depletion of the known and unknown uranium resources by the year 2050. This is similar to the once-thru LWR scenario because of the large LWRs capacity contribution up to this point in time (by the year 2050, about 60% of the total capacity is generated by the LWRs). By the year 2100, the new U discovery needed and the generated enrichment tails are about the same as the corresponding once-thru LWR scenario values. It was expected that a reduction in the required U resources will be achieved as the RTFs are introduced into the market (about 4000 kg U is needed per year for a RTF compared to 23000 kg needed for a LWR). However, the large enrichments associated with the U part of the fuel (~20% in RTF seed compared to ~4% in the LWR fuel) have offset the reduction in the U requirements for the reactors.

As shown in Figure 1, SF from the metallic seed fuel and the ThO2UO2 blanket fuel increase gradually with the introduction of the RTFs. However, by the year 2100, the contributions of those two types of fuel to the total amounts of SF are small compared to the LWR UO2 SF share. In general, the total SF from the RTF scenario is smaller than the SF from the once-thru LWR. Not much reduction in the amount of SF will be achieved by the year 2050, and about 30% reduction will be achieved by the year 2100. This is a result of a much smaller annual discharge of SF from RTF (about 8000 kg which include the Th in the blanket) compared to the LWRs annual discharge (about 23000 kg).

Figure 2 shows the total enrichment and fuel fabrication rates associated with the scenario. The total enrichment rate is higher than the once-thru LWR scenario rate, and it exceeds that rate by almost 30% by the year 2100. Again, this is a result of the higher load imposed on the enrichment plants by the highly enriched RTF fuel. In the other hand, the fabrication rate at the end of simulation has decreased by about 50%. This is a result of the smaller amounts of fuel to be fabricated for the RTF compared to the LWRs. The figure also shows the market penetration of the RTF type reactors compared to the LWRs. By the year 2100, almost all power generation will be attributed to the RTFs after reaching a 50/50 park share by the year 2055.

The different cost estimates for this scenario are shown in Figure 3. The behavior of those individual cost parameters and the total fuel cycle cost per TW-h generated, are similar to the behavior of the costs associated with the once-thru LWR scenario. At the end of simulation (year 2100) the total RTF scenario cost is about the same as the base scenario cost. Notice that the amounts of SF generated by the RTF scenario reduce the storage and shipping costs; however the contributions of those costs to the total cost are very small.

Figure 3 also shows the normalized indexes for this scenario. As expected, from the previous discussion, the cost and ore indexes will not change. The SF index is about 0.7 which correspond to the 30% reduction by the year 2100 in SF, as mentioned before. Notice that the MA fraction in the SF used here was assumed to be the same as that for LWR. This leads to a MA index that is the same as the SF index, i.e., MA index is about 0.7 by the year 2100. The Pu index is about 0.6 showing a substantial reduction in the SF content of Pu. However, as shown in Figure 4, substantial amounts of U233 are also generated in the SF. Although those amounts of U233 are substantial, the U233 in SF is denatured by the existing blanket uranium. Finally, Figure 4 also shows the amounts of mined Th ore. Those amounts are much less than the mined U ore, which suggests a much smaller contribution of the Th ore mining cost to the overall mining cost.

References:

1. A. Galperin, P. Reichert, and A. Radkowsky, “Thorium Fuel for Light Water Reactors – Reducing Proliferation Potential of Nuclear Power Fuel Cycle,” Science & Global Security, 1997, Vol. 6, pp. 265-290, Princeton University.

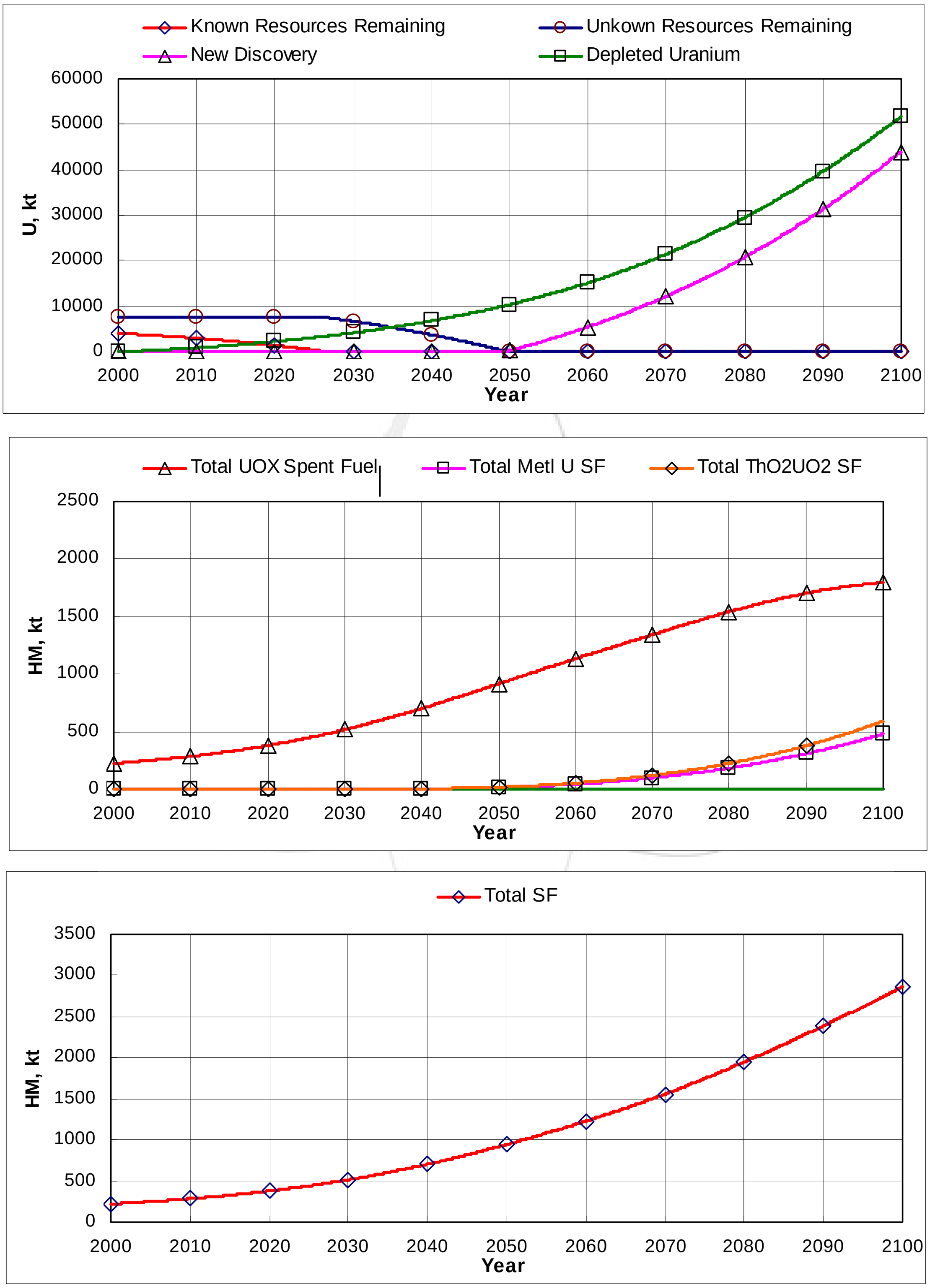
2. A. G. Morozov, A. Galperin, and M. Todosow, “A thorium-based fuel cycle for VVERs&PWRs – a nonproliferative solution to renew nuclear power,” Nuc Eng & Design, pp. 13-14, January 1999.
3. A. Radkowsky and A. Galperin, “The Nonproliferative light Water Thorium Reactor: A new Approach to Light Water Reactor Core Technology,” Nucl. Tech., Vol. 124, Dec. 1998.
4. A. Radkowsky, “Using Thorium in a Commercial Nuclear Fuel Cycle: How to do it,” Nuc Eng & Design, pp. 14-16, January 1999.

Table 1. RTF Reactor Attributes

Power Rating, MWth	3000			
η, %	33			
Load Factor, %	85			
	Seed		Blanket	
Refueling Interval, yr	1		10	
# Batches	3		1	
Burnup, GWd/t	54[a]		100	
kg, HM/yr	Seed		Blanket	
	Input	Output	Input	Output
HM	3625[b]	3206	4450	4450
Total U/yr	3625	2689	445	390
Thorium	0	0	4005	3819
U235/yr	725	128.4	89	5.46
Pu	0	36.6	0	11.8
U233	0	0	0	63.5
FP	0	475.6	0	222.5
MA[c]	0	4.809	0	6.675

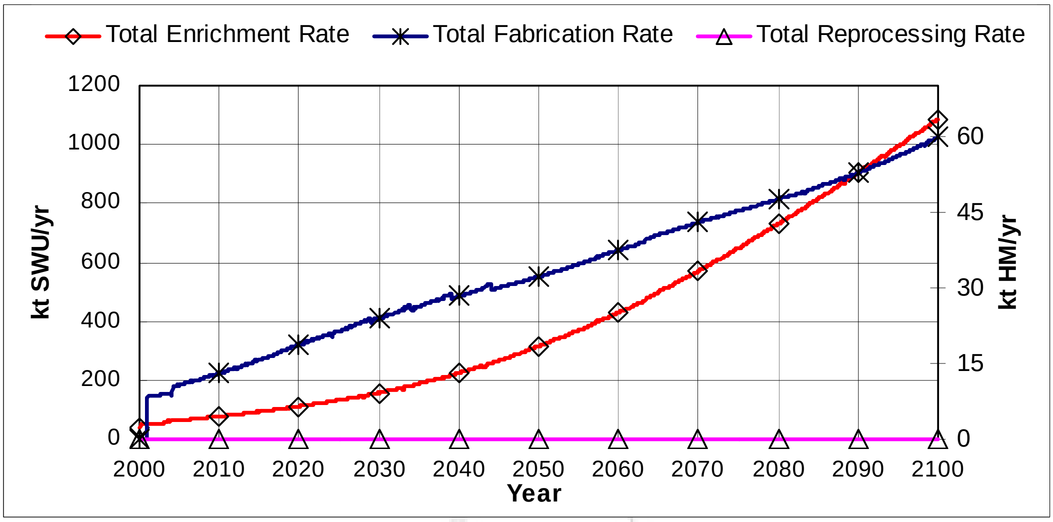
a - Annual burnup
b- An average over cycles 2-9. Initial core loading is 6900 kg HM.
c- Assumed the same minor actinides fractions as LWR (~0.0015 HM)

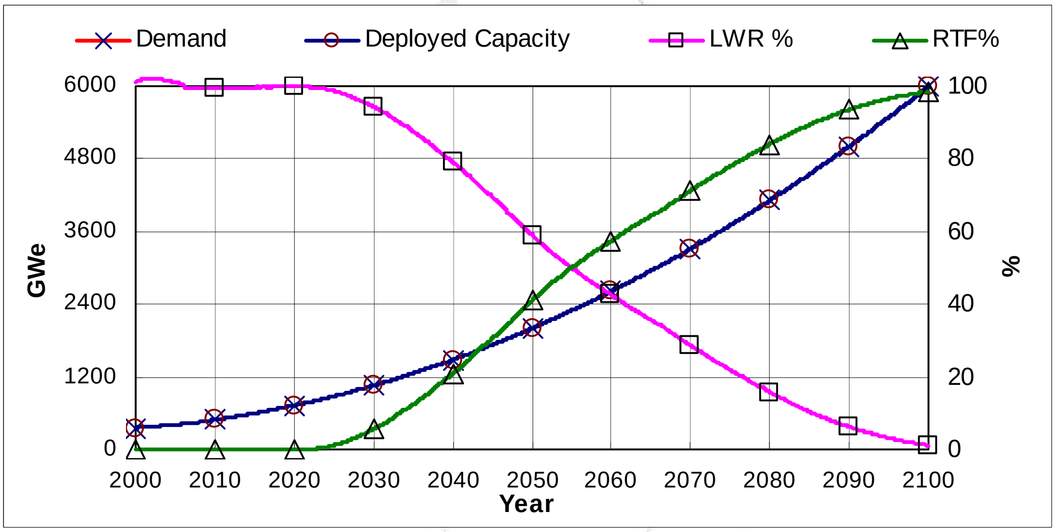
Figure 1



Known Resources Remaining
Unkown Resources Remaining
New Discovery
Depleted Uranium
U, kt
60000
50000
40000
30000
20000
10000
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
Total UOX Spent Fuel
Total Metl U SF
Total ThO2UO2 SF
HM, kt
2500
2000
1500
1000
500
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year
Total SF
HM, kt
3500
3000
2500
2000
1500
1000
500
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

Figure 2


Total Enrichment Rate
Total Fabrication Rate
Total Reprocessing Rate
kt SWU/yr
kt HM/yr
1200
1000
800
600
400
200
0
60
45
30
15
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year


Demand
Deployed Capacity
LWR %
RTF%
GWe
%
6000
4800
3600
2400
1200
0
100
80
60
40
20
0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

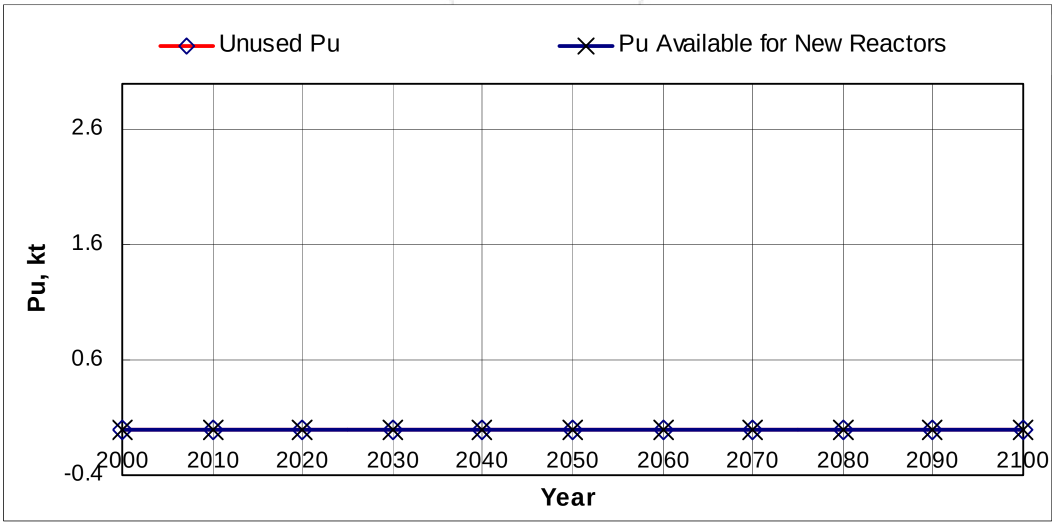
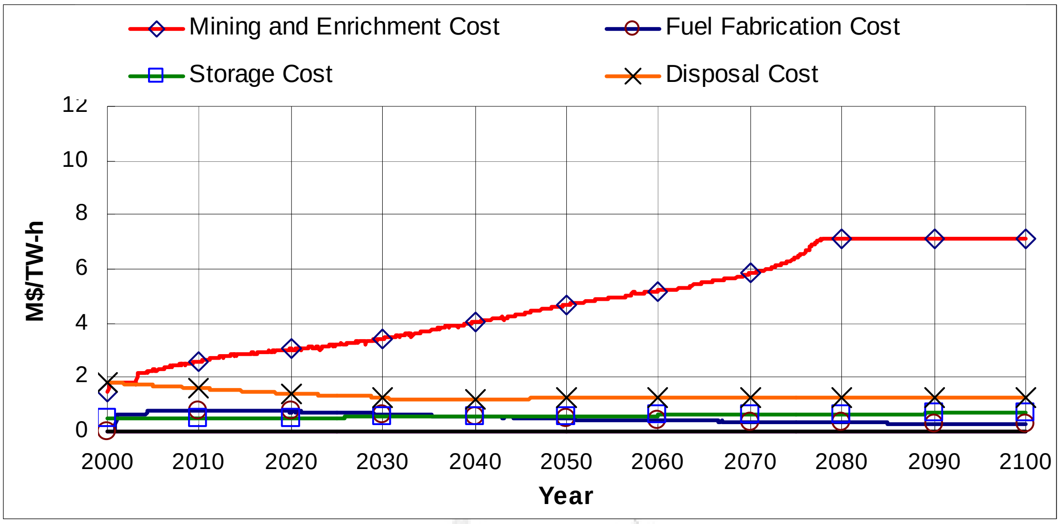
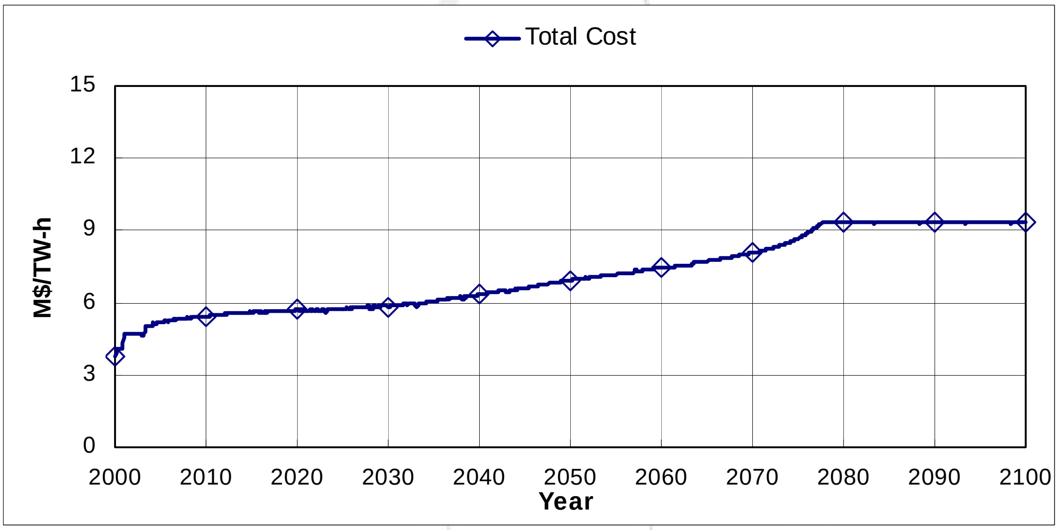

Unused Pu
Pu Available for New Reactors
Pu, kt
2.6
1.6
0.6
-0.4
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

Figure 3.


Mining and Enrichment Cost
Fuel Fabrication Cost
Storage Cost
Disposal Cost
M$/TW-h
Year


Total Cost
M$/TW-h
Year

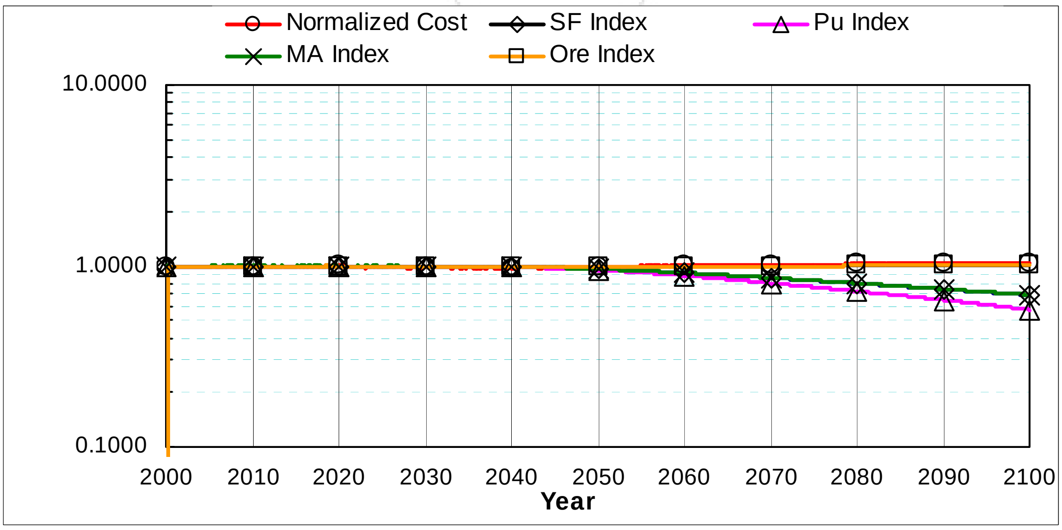
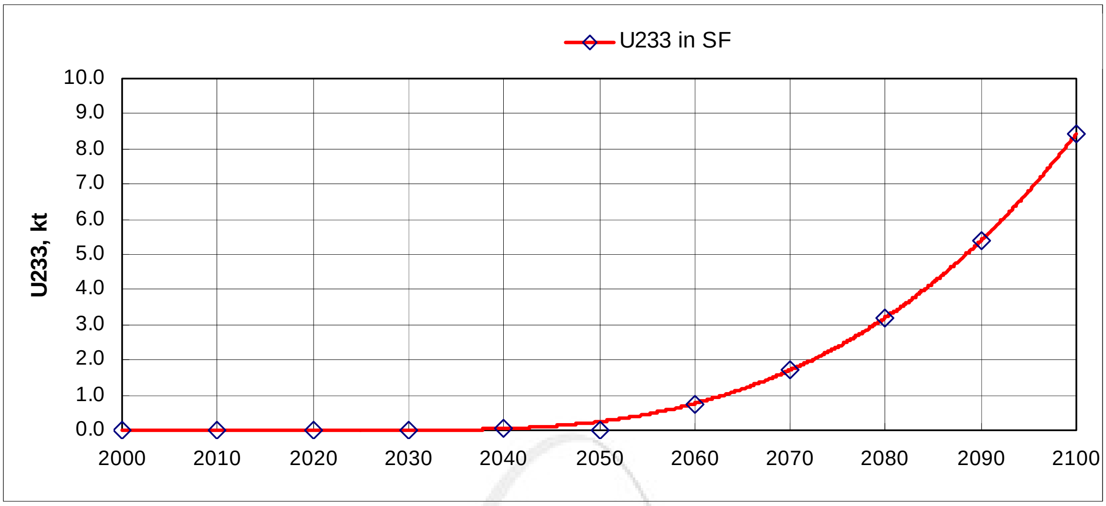
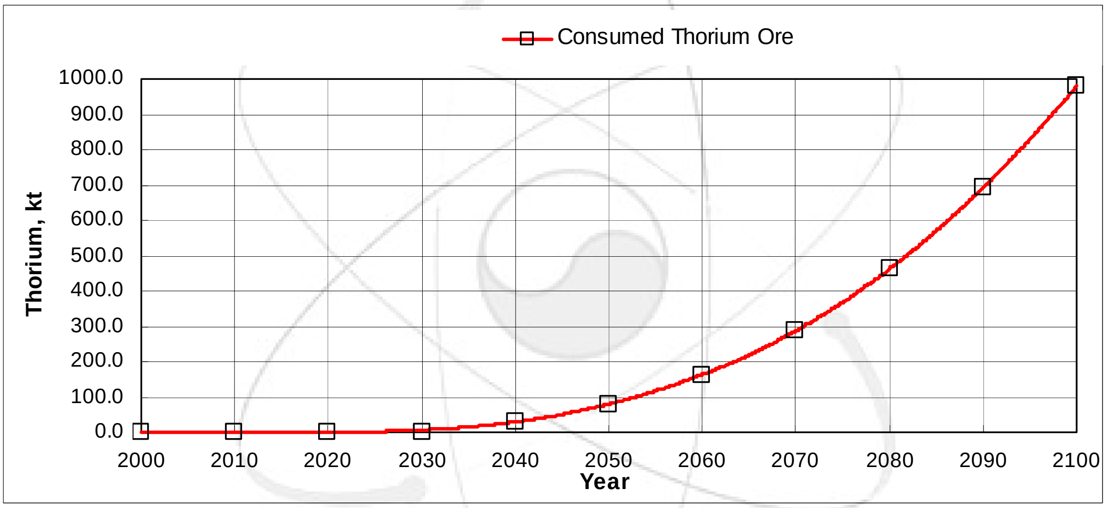

Normalized Cost
SF Index
Pu Index
MA Index
Ore Index
Year

Figure 4.



U233 in SF
U233, kt
10.0
9.0
8.0
7.0
6.0
5.0
4.0
3.0
2.0
1.0
0.0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100



Consumed Thorium Ore
Thorium, kt
1000.0
900.0
800.0
700.0
600.0
500.0
400.0
300.0
200.0
100.0
0.0
2000
2010
2020
2030
2040
2050
2060
2070
2080
2090
2100
Year

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(TR, AR인 경우 주저자)</td><td colspan="4">박주환 (건식공정핵연료기술개발부)</td></tr>
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<tr><td colspan="6">초록 (15-20 줄내외)</td></tr>
<tr><td colspan="6">DYMOND code 는 다이내믹 해석 도구인 ITHINK 플래폼상에서 계산되는 프로그램으로 전세계 에너지 수요에 대해 핵주기 시나리오를 가상하여 향후 100 년 동안의 다이나믹 현상을 평가하기 위해 개발되었다. DYMOND code 코드는 초기에 미국 ANL(Argonne National Laboratory)에서 개발되었으며, LWR once-through 및 LWR-FBR mixed plant 의 해주기 분석에 적용하였으며, 최근에 DYMOND code 의 다른 핵주기 분야에 응용을 위해 수정 보완되었으며, 수정 보완된 DYNOND 는 DUPIC, MSR and RTF 등의 핵주기 분석에 적용할 수 있도록 하였다.
DYMOND code 는 프로그램부, 입력부, 출력부 등 세개의 부분으로 나누어 지지만 이들 각 프로그램부들의 구분이 분명하지 않다. 본 보고서는 수정된 DYMOND code 내의 모형화된 모든 방정식들을 기술하였으며, 다섯개의 부분으로 나뉘어진다.
Part A 는 필요한 핵연료량 및 사용후 핵연료량을포함한 원자로 수명과 관련된 모형을 기술하였고, Part B 는 핵주기 초기 부터 말기동안의 핵연료 흐름에 대한 모형을 기술하였다. Part C 는 연소 우라늄, 플로토늄, Minor Actinide, 저장 및 처분되는 사용후핵연료량 뿐만아니라 핵분열생성물 량등에 대한 모형을 다루었다. Part D 는 MSR 및 RTF 와 같은 토륨 핵주기를 다루는 다른 핵주기에 대한 모형을 Part E 는 경제성에 대한 모형을 다루었으며, 우라늄 채광 비용, 원자로 운전비용, 핵연료 비용등에 대한 정보를 기술하였다.다.</td></tr>
<tr><td colspan="6">주제명키워드(10 단어 내외)</td></tr>
<tr><td colspan="6">DYMOND, DUPIC, Fuel Cycle, GEN-IV</td></tr>
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<tr><td colspan="2">Performing Org. Report No.</td><td colspan="2">Sponsoring Org. Report No.</td><td colspan="2">Standard Report No.</td><td colspan="2">INIS Subject Code</td></tr>
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<tr><td>Title/Subtitle</td><td colspan="7">MODELING REPORT of DYMOND Code (DUPIC version)</td></tr>
<tr><td>Project Manager</td><td colspan="7">Joohwan Park,
Dry-processed Nuclear Fuel Technology Development Division</td></tr>
<tr><td>Researcher and Department</td><td colspan="7">Abdellatif M. Yacout (Argonne National Laboratory)</td></tr>
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Abstracts (15-20 Lines)

The DYMOND code employs the ITHINK dynamic modeling platform to assess the 100-year dynamic evolution scenarios for postulated Global Nuclear Energy Parks. Firstly, DYMOND code has been developed by ANL(Argonne National Laboratory) to perform the fuel cycle analysis of LWR once-through and LWR-FBR mixed plant. Since the extensive application of DYMOND code has been requested, the first version of DYNOND has been modified to adapt the DUPIC, MSR and RTF fuel cycle.

DYMOND code is composed of three parts; the source language platform, input supply and output. But those platforms are not clearly distinguished. This report described all the equations which were modeled in the modified DYMOND code (which is called as DYMOND-DUPIC version). It divided into five parts;

Part A deals Model in Reactor History which is included amount of the requested fuels and spent fuels. Part B aims to describe Model of Fuel Cycle about fuel flow from the beginning to the end of fuel cycle. Part C is for Model in Re-processing which is included recovery of burned uranium, plutonium, minor actinide and fission product as well as the amount of spent fuels in storage and disposal. Part D is for Model in Other Fuel Cycle which is considered the thorium fuel cycle for MSR and RTF reactor. Part E is for Model in Economics. This part gives all the information of cost such as uranium mining cost, reactor operating cost, fuel cost etc..

Subject Keywords (About 10 words)

DYMOND, DUPIC, Fuel Cycle, GEN-IV