Study on the sensitivity analysis of the installed capacity and the high-level waste generation based on closed nuclear fuel cycle

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Abstract：The sustainable development of nuclear energy calls for the maximization of uranium utilization and, meanwhile, the minimization of the waste produced. The fuel cycle mode and its key parameters have a great influence to the deployment of nuclear energy and the generation of high-level waste. In this paper, a sensitivity analysis is made on the influence of out-of-core residence time and the recovery ratio of the actinides; a series of recommended value are given to the parameters mentioned above; the effect of the closed fuel cycle on the reduction of high-level waste is analyzed. The following conclusions are drawn: 1) in the multi-cycle of industrial Pu or transuranics(TRU) in fast reactors, the contents will reach an equilibrium state; for the CFR1000, of which the breeding ratio is 1.2, the fissile Pu takes an fraction of 70% in the equilibrium state, and the minor actinides(Mas) take an fraction of 3% in the TRU recycle; 2) the installed capacity of fast reactor is very sensitive to the out-of-core residence time and recovery ratio; the generation of high-level waste is sensitive to the recovery ratio; the recommendation is that in the future development of the fuel cycle, the out-of-core residence time be no more than 5 years and the recovery ratio of actinides be no less than 99.9%; the reasons are that in order to avoid the decreasing of the nuclear installed capacity during the transition from PWRs to FRs, the out-of-core residence time is proposed to be no more than 5 years and the recovery ratio be no less than 99%, and that in order to reduce the recycling loss and decrease the generation of the waste, the recovery ratio is proposed to be no less than 99.9%; the benefit of further improvement of the ratio beyond 99.9% is insignificant either for the installed capacity or for the high-level waste; 3) the synergistic development of the PWRs and FRs in closed fuel cycle can not only improve the utilization of uranium but also effectively reduce the generation of the high-level waste; compared with the once-through method in PWRs, closed-fuel-cycle can reduce the long-term radioactive toxicity of high-level waste to 1/5~1/6 with a recovery ratio of 99.9%; the TRU whole cycle can effectively reduce the amount of MAs and further reduce the long-term radioactive toxicity of high-level waste to 1/7~1/8.

Keywords：Out-of-core residence time; recovery ratio of actinides; nuclear power installed capacity; high-level waste generation;

1. Introduction

The key to sustainable development of nuclear energy is to maximize U resource utilization and to minimize waste (especially high level waste) generation. The nuclear fuel cycle has two types of once through and closed fuel cycle, in which the closed fuel cycle can greatly increase the utilization rate of natural U resource. To achieve closed fuel cycle, spent fuel reprocessing, fresh fuel manufacturing and advanced reactors are three indispensables. Fast reactor, which can realize effective fuel breeding and actinide waste transmutation, is the key part of nuclear system with closed fuel cycle. Considering whether the nuclear system with fast reactor and closed fuel cycle system can achieve breeding and transmutation target, the key performance is the total installed capacity of Nuclear Power Plant (NPP) in case of limited natural U resource and the total generation of high level waste, while the former reflects the developable scale of nuclear energy and the latter determines the demand to final geological repository. As the actinides is the most important part of high level waste in long time scale, in this paper high level waste focuses on the actinides.

The design and operation parameters of spent fuel reprocessing and fresh fuel manufacturing will affect the whole nuclear energy system performance. Focusing on the two parameters of fuel out-pile residence time and actinides recovery rate in closed fuel cycle, the sensitivity of which to the installed NPP capacity and the generation of high level actinides waste is studied, and the recommended requirements are proposed, in the paper, which will provide a reference for the development of nuclear fuel cycle in the future.

2. Computation tools and reactor selection

2.1.Computation tools

The fast reactor core neutronics and fuel burnup calculation uses CITATION code[1] and PASC-1 code system[2], which generates few group cross-sections for CITATION code.

The fine 171 groups NVitamin-C[3] library is used as the source library. This library is an updated version of Vitamin-C library and developed by China Institute of Atomic Energy based on the evaluated nuclear data libraries ENDF/B-VI, JEF-2, CENDL-2 and JENDL-3. Compared with the Vitamin-C library, the new library contains more nuclides (from 66 nuclides to 105 nuclides, two of them are pseudo fission products of 235U and 239Pu, respectively) and is processed by newer evaluation library.

The few group microscopic cross-section library prepared for CITATION code is generated by PASC-1 code system for the specific core layout of geometry. PASC-1 code system is a code package for collapsing multi-group cross section into few groups for CITATION code. The CITATION-used cross section library can be processed by a variety of ways, but XSDRN code[4], 1D SN transport code in PASC-1 code system, is specifically designed for this purpose.

Full core 3D diffusion calculation, including steady state and burnup, is performed by CITATION code, which is widely used in reactor core neutronics analysis and is proven to be reliable.

The calculation of fuel mixture composition during reprocessing is performed by the newly developed MIX code, which can simulate the mixing of spent fuel and blanket according to different reprocessing ways, and calculate the composition of recovered Pu or transuranics (TRU) and make up the fresh fuel, and then coupled with CITATION code to do fuel burnup calculation.

The self-developed FCA code is used to do dynamic simulation and analysis of the closed nuclear fuel cycle. The FCA code can track all important actinides and several fission products and calculate the mass balance of these materials according to the selected reactor type and fuel cycle parameters.

2.2.Reactor selection

Two types of reactors are selected in the analysis, one is pressurized water reactor (PWR) and the other is sodium cooled fast reactor (SFR). For PWR, the M310 reactor from Daya Bay NPP with electricity output of 1000MW, refueling period of 1 year and the average fuel discharge burnup of 33000MWd/t is choose; for SFR, the self-designed CFR1000 with electricity output of also 1000MWe, refueling period of 160days and the average fuel discharge burnup of 60600MWd/t is choose. There are two ways for SFR fuel recycle, one is Pu recycle and the other is TRU recycle.

The main parameters for different reactors are shown in TABLE 1.

TABLE 1 Main parameters for typical PWR and SFR

	M310[5]	CFR1000[6]
Electricity power/MW	1000	1000
Lifetime/a	60	60
Refueling period/EFPDs	310	160
Fuel type	UO2	MOX(Industrial Pu and depleted U)
Average fuel discharged burnup /(MWd/tHM)	33000	60600
HM loading/t	72.5	51.3
In which: fuel active zone/t	72.5	18.9
axial blanket/t	-	14.5
radial blanket/t	-	17.9
Fuel enrichment	3.15%	16.0%(Inner zone) /18.2%(Middle zone) /20.3%(Outer zone)①
Composition of discharged fuel/wt	U：95.6%； Pu：1.1%； MA (Minor actinides): 0.1%； FP (Fission product）: 3.2%	Calculated by MIX and CITATION code

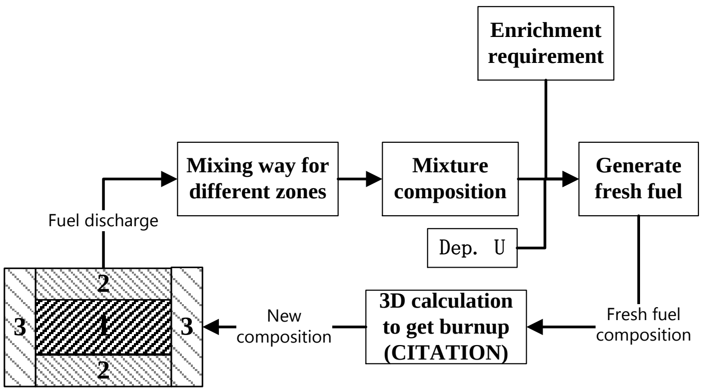
①：which means PuO2 weight percentage in MOX fuel.

3. Equilibrium composition of fast reactor fuel

Industrial Pu or TRU, recovered from PWR or SFR spent fuel, can be utilized as loading fuel in SFR, and be recycled for many times. After each cycle, when do fuel reprocessing, assumes that the discharged driving fuel and blanket will be reprocessed together. For TRU recycle, during reprocessing, the fission products and partial U will be separated and TRU (Pu + MA) plus a certain amount of U will be recovered as one group to ensure no mass stream of pure Pu in the reprocessing flow, and then the recovered TRU will be further multi-recycled in reactors (like CFR1000). To realize this, some new separation technologies have been developed, such as COEX which produces a U-Pu blended product[7], and NUEX which yields a U-Pu-Np blend[8], and UREX+1a which separates TRU as one group[9], or combined aqueous and pyrochemical processes to recover all TRU elements together[10].

After multi-recycled, the Pu or TRU composition will change and reach the equilibrium state [11], which will affect SFR spent fuel composition and finally affect the actinides composition in nuclear fuel cycle. Therefore, it needs to determine the Pu or TRU composition changing characteristics quantitatively.

The SFR core is usually designed with active zone, axial and radial blanket, and for the spent fuel from each zone, the reprocessing can be combined according to different needs. Thus, the MIX code, to do mixing and coupling with CITATION code, is developed, and the



calculation flow is shown in

FIG *1*.

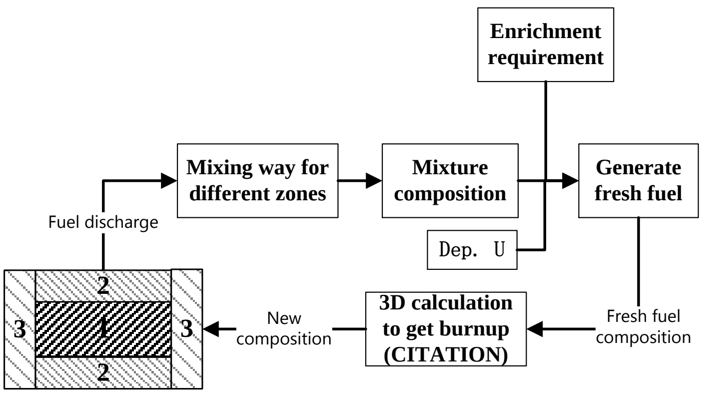


FIG 1   Calculation flow for MIX code

The equilibrium composition for Pu or TRU recycled in CFR1000 is shown in TABLE 2, which will be used as input for the following nuclear fuel cycle analysis.

TABLE 2 The normalized composition for equilibrium state

	Pu recycle	TRU recycle
^{238}Pu	1.68E-04	1.65E-04
^{239}Pu	7.24E-01	7.08E-01
^{240}Pu	2.49E-01	2.38E-01
^{241}Pu	1.82E-02	2.32E-02
^{242}Pu	8.80E-03	9.51E-03
^{237}Np	-	2.75E-03
^{241}Am	-	1.39E-02
^{243}Am	-	1.87E-03
^{243}Cm	-	5.15E-05
^{244}Cm	-	1.55E-03
^{245}Cm	-	3.68E-04
SUM	1.00	1.00

4. Sensitivity analysis of the influence of key fuel cycle parameters on the installed capacity and the high-level waste generation

In the hypothetical nuclear system with closed fuel cycle, PWR and SFR are important parts. The PWRs will provide initial industrial Pu for SFR startup, and the SFRs can gradually expand the installed capacity of nuclear power by breeding after startup. Selecting the installed capacity and the generation of actinides waste as the target respectively, the influence of key fuel cycle parameters, such as fuel out-of-core residence time and recovery ratio of actinides, on the above-mentioned target is studied in the paper.

4.1. Calculation method

4.1.1. Nuclear system simulation

The actinides and fission products in the key parts of closed fuel cycle such as nuclear power plant, spent fuel reprocessing plant, fuel fabrication plant, intermediate storage and final geological repository should be tracked and do balance calculation.

The newly developed FCA code could simulate the closed nuclear fuel cycle system dynamically, as shown in FIG 2, and track each nuclide mass. The simulated parameters include fuel out-of-core residence time and recovery ratio.

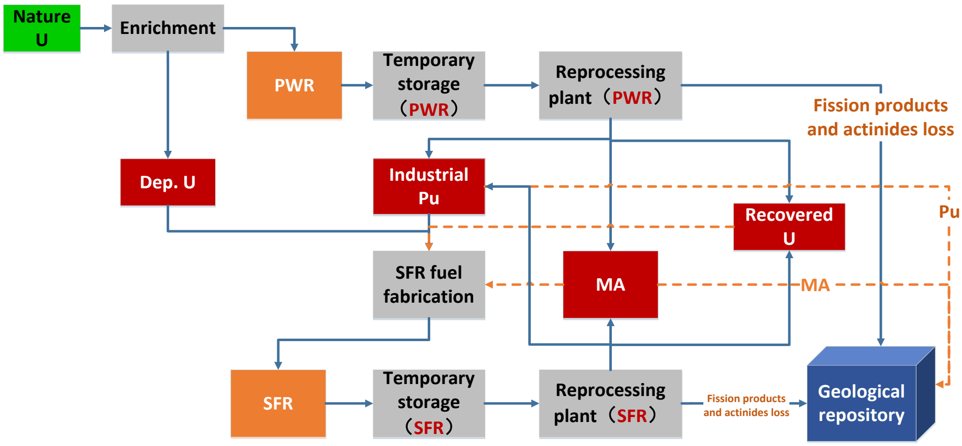


FIG 2 The closed fuel cycle system simulated by FCA code

4.1.2. System doubling time

The FCA simulation assumes that, for each fuel cycle once the spent fuel is reprocessed, the recovered Pu or TRU would be stored and can be used to start a new SFR. In this scenario, the fissile material and the corresponding installed SFR capacity in the system will increase exponentially, and the system doubling time[12] could be used to describe the nuclear system capacity increasing performance. If the installed SFR capacity follows the following formula:

$$N(t) = N_0 e^{\lambda t} \qquad (1)$$

In which, N means the installed SFR capacity at time t. Thus, the system doubling time is defined as $\frac{ln2}{\lambda}$.

4.2.The installed capacity in nuclear system

4.2.1. The assumptions

The M310 type PWR and the CFR1000 type SFR will be deployed in the nuclear system shown as FIG 2. The main assumptions made for quantitatively analysis include:

1) The retrievable natural U resource is 2 million tones and can support 200GWe PWR development in China, and the development rate is assumed to be 5GWe/a;

2) Only Pu is recovered and recycled, MA is dropped directly to high level waste; spent fuel discharged from PWR and SFR will be reprocessed immediately when available, and do not consider the reprocessing capacity limit; once the Pu reserves can meet the first loading requirement and the following refueling requirement, CFR1000 would be constructed immediately, and do not consider the construction capacity limit;

3) The out-of-core fuel storage time consider 4, 6, 8, 10 and 12 year, and the recovery ratio consider 95%, 99%, 99.7%, 99.9% and 99.99% in our analysis; when do the out-of-core fuel storage time sensitivity analysis, the recovered rate keeps fixed and selects 99.9% for U and Pu, and when do the recovery ratio sensitivity analysis, the out-of-core fuel storage time keeps 5 year;

4) The simulated time scale is 200 years.

4.2.2. The influence of out-of-core residence time

4.2.2.1. The sensitivity difference of out-of-core residence time of PWR and SFR fuel

Because of the difference of spent fuel reprocessing and fabrication technology between PWR and SFR, the fuel out-of-core residence time in the nuclear fuel cycle may be different. Therefore, the sensitivity difference of out-of-core residence time between the two types of reactors is studied at first, and the calculation results of installed nuclear capacity are shown in FIG 3 and FIG 4. The results indicate that the PWR fuel out-of-core residence time will introduce little influence on SFR capacity, while the SFR fuel out-of-core residence itself will have a big impact on SFR capacity. Based on this, the subsequent analysis assumes that PWR and SFR has the same fuel out-of-core residence time.

4.2.2.2. The influence of fuel out-of-core residence on nuclear system installed capacity

Driven by the industrial Pu recovered from the spent fuel discharged from 200GWe PWR, and based on different fuel out-of-core residence time, the development of CFR1000 installed capacity is shown in FIG 5, and the system doubling time is shown in TABLE 3.

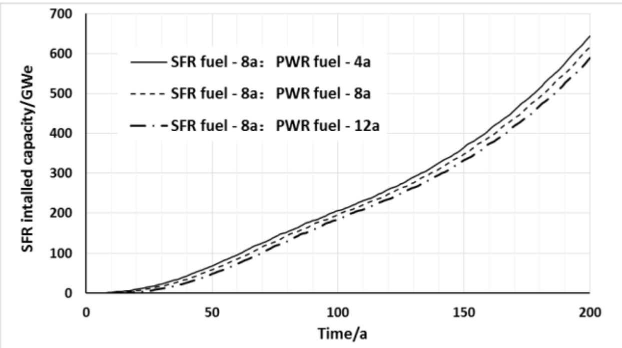


FIG 3 The influence on the installed capacity caused by PWR spent fuel out-of-core residence time (assuming SFR fuel out-of-core residence time keeps unchanged)

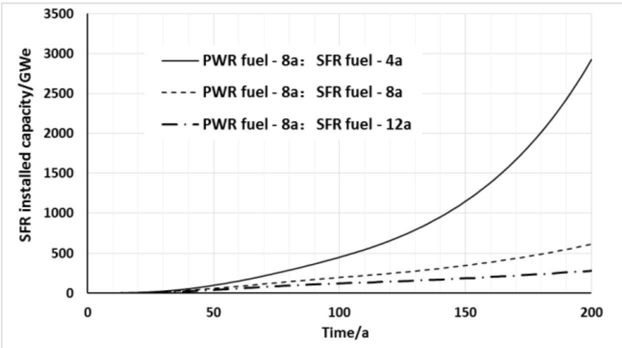


FIG 4 The influence on the installed capacity caused by SFR spent fuel out-of-core residence time (assuming PWR fuel out-of-core residence time keeps unchanged)

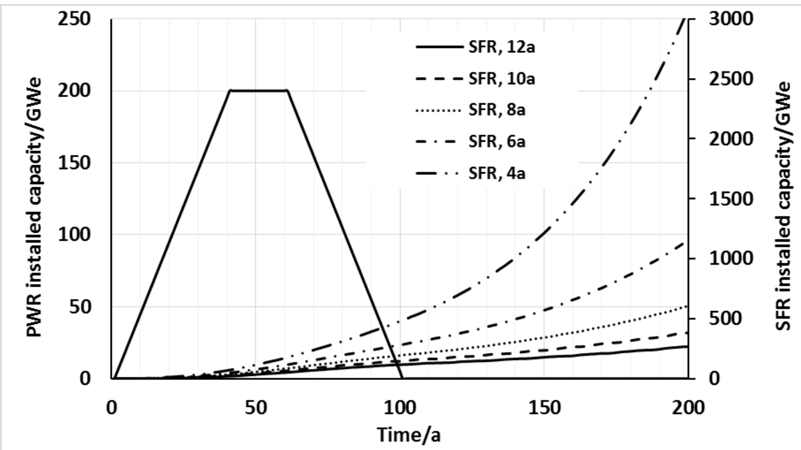


FIG 5 The development of SFR installed capacity with different fuel out-of-core residence time (the recovery ratio is 99.9%)

TABLE 3 The system doubling time for different fuel out-of-core residence time (the recovery ratio is 99.9%)

Fuel out-of-core residence time/a	4	6	8	10	12	8(the breeding ratio increased to 1.5)
The system doubling time/a	37.1	49.0	60.8	72.5	84.2	46.9

The calculation results indicate that, the fuel out-of-core residence time consumed by fuel reprocessing and refabrication will bring significant influence on the system doubling time and the system installed capacity. For example, for CFR1000 reactor, if the fuel out-of-core residence time is increased from 4a to 12a, the related system doubling time would increase from 37a to 84a, and the system installed capacity difference in 200 years will be of about 10 times.

4.2.2.3. SFR development mode analysis

FIG 6 and FIG 7 show the development of SFR installed capacity with fuel out-of-core residence time of 8a and 5a respectively, in which it can be concluded that the typical matching development of PWR and SFR could be divided into two stages, SFR startup stage and SFR developed in large scale stage. In SFR startup stage, PWRs start to be deployed and provide Pu for SFR startup, and the stage is ended when all PWRs are closed; in SFR developed in large scale stage, PWRs are all closed and the system capacity completely relies on SFR, and the system doubling time reflects the growth rate of nuclear system installed capacity in this stage.

In SFR startup stage, the system capacity is affected greatly by PWRs, along with the gradually closing of PWR, if the increasing SFR capacity is slow, the total capacity of nuclear system may decrease as shown in FIG 6, which is unfavorable for nuclear energy development. If nuclear energy is used as the basic energy, it is hoped that the total installed capacity of nuclear power plants (NPP) will rise steadily, and gradually reach the target size, and then remain stable.

Therefore, from view of this point, it needs to raise the requirement to fuel out-of-core residence time, to ensure that the total installed capacity of NPP will not decrease during the transition from PWRs to SFRs. In this study, when uses M310 PWR and CFR1000 SFR, and assumes that the maximum PWR capacity is 200GWe and the recovery ratio for PWR and SFR spent fuel is 99.9%, the fuel out-of-core residence time no longer than 5a can achieve the target of not reducing of the total NPP installed capacity during the transition, just as shown in FIG 7.

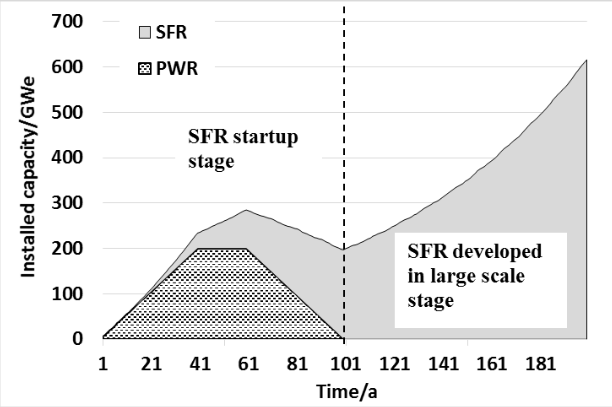


FIG 6 The development of SFR installed capacity with fuel out-of-core residence time of 8a (the recovery ratio is 99.9%)

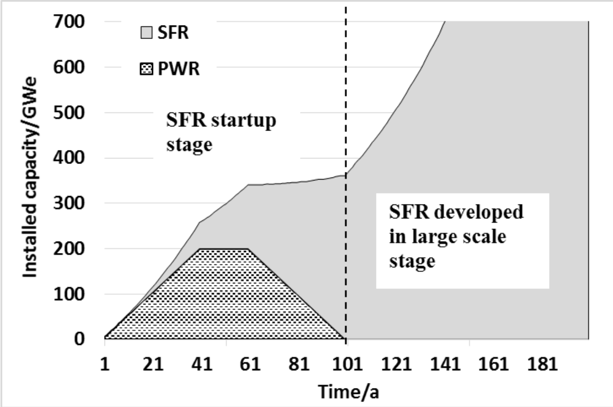


FIG 7 The development of SFR installed capacity with fuel out-of-core residence time of 5a (the recovery ratio is 99.9%)

4.2.3. The influence of actinides recovery ratio

4.2.3.1. The sensitivity difference of out-of-core residence time between PWR and SFR fuel

The recovery ratio of PWR spent fuel reprocessing will introduce little influence on SFR capacity, while the SFR fuel recovery ratio itself will have a big impact on SFR capacity, as shown in FIG 8 and FIG 9. Based on this, the subsequent analysis assumes that PWR and SFR has the same spent fuel recovery ratio.

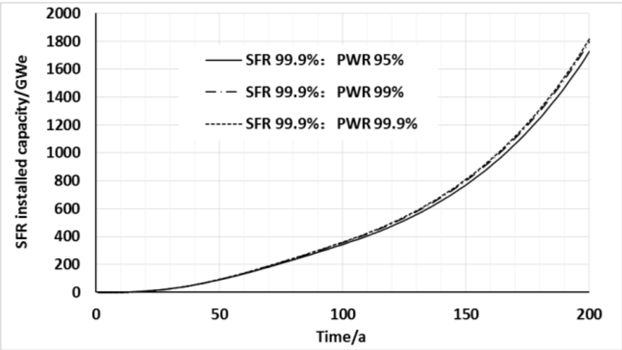


FIG 8 The influence on the installed capacity caused by PWR spent fuel recovery ratio (assuming SFR spent fuel recovery ratio keeps unchanged)

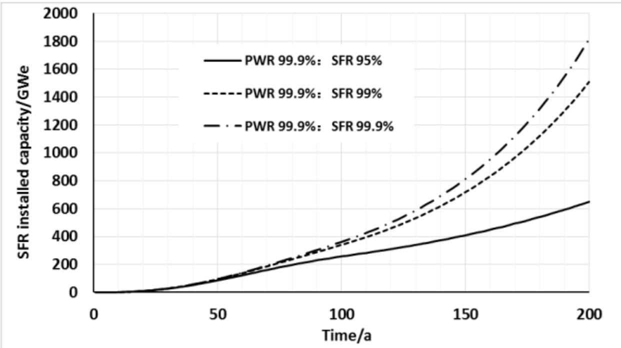


FIG 9 The influence on the installed capacity caused by SFR spent fuel recovery ratio (assuming PWR spent fuel recovery ratio keeps unchanged)

4.2.3.2. The influence of recovery ratio on nuclear system installed capacity

Driven by the industrial Pu recovered from the spent fuel discharged from 200GWe PWR, and based on different fuel recovery ratio, the development of CFR1000 installed capacity is shown in FIG 10, and the corresponding system doubling time is shown in TABLE 4.

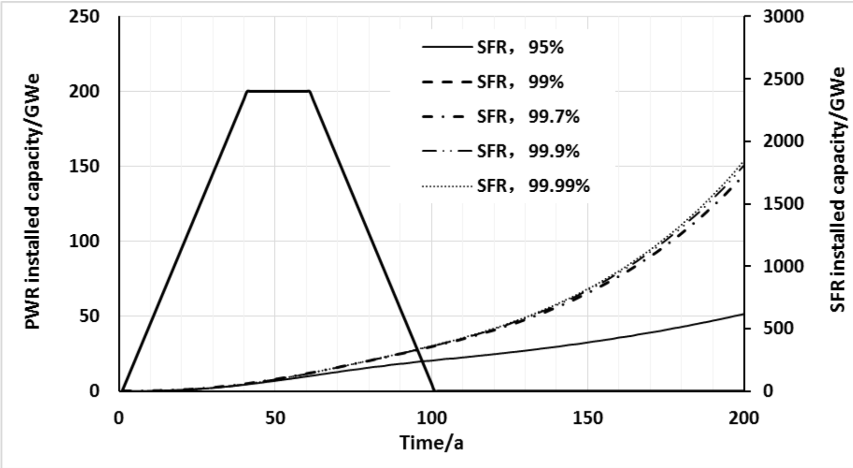


FIG 10 The development of SFR installed capacity with different recovery ratio (the fuel out-of-core residence time is 5a)

TABLE 4 The system doubling time for different recovery ratio (the fuel out-of-core residence time is 5a)

Recovery ratio	95%	99%	99.7%	99.9%	99.99%
The system doubling time/a	75.2	46.7	43.8	43.1	42.7

The calculation results indicate that:

1) The recovery ratio in spent fuel reprocessing will bring significant influence on the system doubling time and the system installed capacity. For example, for CFR1000 reactor, if the recovery ratio is increased from 95% to 99%, the related system doubling time would decrease from 75a to 46a, and the system installed capacity difference in 200 years will be of about 3 times.

2) After the recovery ratio is increased to 99%, then the further increase of it has limited effect to decrease the system doubling time and to increase the system installed capacity, but has great effect to waste generation, which will be described in detail in the subsequent section.

4.2.3.3. The influence of recovery ratio on SFR development mode

Similar to section 4.2.2.3, from the view-point of ensuring the total NPP installed capacity not reducing during the transition from PWRs to SFRs, it also needs to raise the requirement to recovery ratio. In this study, when uses M310 PWR and CFR1000 SFR, and assumes that the maximum PWR capacity is 200GWe and the fuel out-of-core residence time is 5a, the fuel recovery ratio no lower than 99% can achieve the target of not reducing of the total NPP installed capacity during the transition, just as shown in FIG 12.

By combining the requirement analysis of fuel out-of-core residence time and recovery ratio, to ensure the total NPP installed capacity not reducing during transition, the technical requirement of no lower than 99% recovery ratio and no longer than 5a fuel out-of-core residence time is proposed, and these parameters will be used in the following waste generation analysis section.

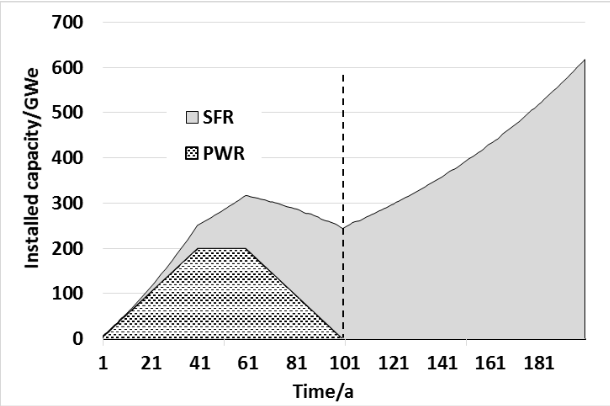


FIG 11 The development of SFR installed capacity with the recovery ratio of 95% (the fuel out-of-core residence time is 5a)

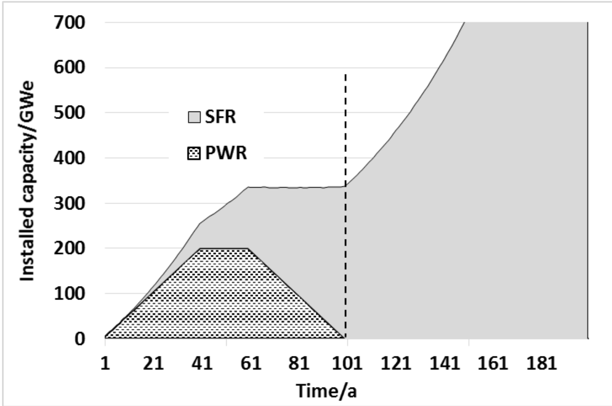


FIG 12 The development of SFR installed capacity with the recovery ratio of 99% (the fuel out-of-core residence time is 5a)

4.3.The actinides waste generation analysis

4.3.1. The assumptions

Also, the M310 type PWR and the CFR1000 type SFR will be deployed in the nuclear system shown as FIG 2. The main assumptions made for quantitatively analysis include:

1) The retrievable natural U is 2 million tones and can support 200GWe PWR development in China, and the development rate is assumed to be 5GWe/a;
2) Because the total NPP installed capacity will affect the waste generation directly, the target NPP installed capacity is assumed to be 400GWe, and the total installed capacity should maintain stable once the target is achieved, which means the new building of NPP is only used to replace the shutdown plants; meanwhile, since there is no need to increase nuclear power, it is assumed that the newly constructed SFR will reduce reactor breeding ratio, only to maintain the fissile materials amount in the system stable;
3) The TRU is recovered and recycled as a group, which means MA and Pu is recovered at same time; the spent fuel discharged from PWR and SFR will be reprocessed immediately when available, and do not consider the reprocessing capacity limit; once the TRU reserves can meet the first loading requirement and the following refueling requirement, CFR1000 would be constructed immediately, and do not consider the construction capacity limit;
4) In order to compare the influence on actinides waste generation after introducing SFR with TRU recycling, the modes of only using PWR with fuel once through cycle and SFR only considering Pu recycling are also additionally included;
5) In the analysis, the fuel out-of-core residence time in FCA code is assumed to be fixed for 5a; the recovery ratio in FCA is taken into account 99% and 99.9% respectively;
6) The simulated time scale is 300 years.

4.3.2. The analysis of actinides waste generation

4.3.2.1. The analyzed scenarios

The assumed target for nuclear energy is that the installed capacity is increased gradually to 400GWe and then keeps stable, and the total operation time for nuclear energy is 300 years. The analyzed problem is the total actinides waste generation of the nuclear energy system.

To meet the target mentioned above, there are several scenarios are analyzed, including the followings.

Scenario A: only PWR with once through fuel cycle is deployed. The cumulated operation of reactor years is assumed to be consistent with the other scenarios with closed fuel cycle, and at the time of the 300th year the total PWR installed capacity is assumed be 400GWe.

Scenario B: PWR is deployed first and SFR is started by using the PWR generated Pu; the fuel cycle is partly closed with only recycle Pu (MA is abandoned directly); after startup, the SFR installed capacity is developed gradually to 400GWe, then maintains the scale and operates to 300 years; the fuel out-of-core residence time is 5a and the recovery ratio is 99.9%.

Scenario C: the scenario is similar to scenario B but with TRU recycle to replace Pu recycle, which means MA are also recycled in SFR.

In different scenarios, the components for actinides waste is shown in TABLE 5.

TABLE 5 The components for actinides waste generation for different scenarios

	The components for actinides waste generation
Scenario A	1) Spent fuel generated during PWR reactors operation in 0-300 year 2) All the fuel in the 400GWe PWR reactors at the 300th year
Scenario B	1) Fuel reprocessing loss during SFR operation in 0-300 year (a little part of U and Pu, and all of MA) 2) All the actinides in 400GWe SFR reactors at the 300th year 3) Pu in the temporary storage
Scenario C	1) Fuel reprocessing loss during SFR operation in 0-300 year (a little part of U, Pu and MA) 2) All the actinides in 400GWe SFR reactors at the 300th year 3) TRU in the temporary storage

4.3.2.2. The calculation results and analysis

（1）The comparison of the installed capacity development

The system installed capacity for scenario B and scenario C is compared in FIG 13. Because there is a little MA percentage in the fuel equilibrium composition in TRU recycle mode, the impact on core breeding performance is insignificant and the increasing of the installed capacity for scenario B and C is almost the same, in which the target 400GWe capacity is achieved at about the 105th year and then keeps stable at this level. After 300 years of operation, the total NPP operation time (including PWRs) for scenario B and C is 104614GWe·a and 105210GWe·a respectively.

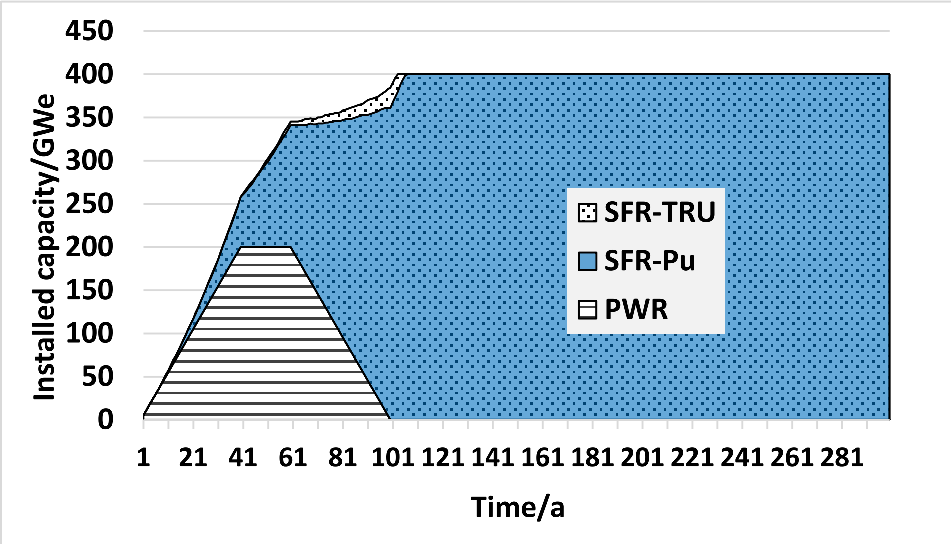


FIG 13 The system installed capacity for scenario B and Scenario C

(the fuel out-of-core residence time is 5a and the recovery ratio is 99.9%)

（2）The comparison of generation of actinides waste

The actinides waste generation for different scenarios is presented in TABLE 6, in which the results indicate:

1) Under the same nuclear energy installed capacity and the same operation reactor years, developing SFR and closed fuel only generates about 100,000 tons waste of actinides and fission products, of which most are fission products, accounting for about 75%, and U is about 20%, and Pu and MA only accounts about 5%; if deploys only PWR with once through fuel cycle, a huge amount of 2,500,000 tons of high level actinides waste will generate, of which 95% is U;

2) PWR with once through fuel cycle will produce a large number of industrial Pu, which is about 10 times of that in the condition of developing SFR with closing fuel cycle, and the cumulative Pu is the main source of long term radiotoxicity in high level waste; SFR can be designed with flexible reactor core breeding ratio, and can maintain the total industrial Pu amount stable in the nuclear energy system;

3) The integrated TRU recycling can effectively reduce and maintain the total MA amount in the nuclear energy system, which is only about 20% of Pu recycling; PWR with once through fuel cycle will generate the most amount of MA, about 1.8 times of that in the conditions of developing SFR with Pu recycling.

（3）The sensitivity analysis of recovery ratio influence on actinides waste generation

Since the recovery ratio directly affects the reprocessing loss in nuclear fuel cycle, it will bring significant influence on actinides waste generation, as shown in TABLE 7, which is calculated under scenario C but the fuel recovery ratio is changed to 99%. With the recovery ratio decreasing from 99.9% to 99%, the corresponding reprocessing loss increases about 10 times, and finally results that at the same nuclear energy generation, the system waste generation increases 30%, about 30,000 tons, in which more than 90% is U loss increasing, and then followed by Pu loss increasing. The benefit to reprocessing loss waste reducing brought by further increasing of recovery ratio to more than 99.9% is very limited.

（4）The characteristics of high level actinides waste for different scenarios

By using ORIGEN2[13] code, the radioactivity, the heat release and the ingestion toxicity of actinides waste generated in different scenarios are calculated and shown in TABLE 8.

The amount of high level waste directly determines the requirement to final geological repository. For comparison, selecting the Yucca mountain repository as the base for analysis, and TABLE 9 gives the relative value of the results, in which each characteristic parameter is normalized to the Yucca mountain repository result. Since the decay half-life for most of fission products is relatively short to Pu and MA and the contribution of fission products is really small after 1000 years, after stored for 1000 years only the actinides characteristics is calculated and compared with the Yucca mountain repository, of which the designed capacity is 70,000 tons of discharged spent fuel from PWR with once through fuel cycle.

The results in TABLE 8 and TABLE 9 indicate:

1) By deploying SFR and closed fuel cycle, the radioactivity, the heat release and the ingestion toxicity of high level waste generated in nuclear system is much lower than that of PWR with once through fuel cycle.
2) Under the assumed nuclear development scale, deploying PWR with once through fuel cycle, the generated high level waste needs about 37 repositories like Yucca mountain to disposal, while deploying SFR and closed fuel cycle, only needs 5-7 repositories like Yucca mountain to disposal, which significantly reduces the requirement to high level waste geological repository.

TABLE 6 The high level generation for different scenarios (only considers actinides and fission products)

Unit/t	Scenario A			Scenario B				Scenario C			
	Generated by operation	Fuel in reactors	Subtotal	Reprocessing loss	Fuel in reactors	Temporary storage	Subtotal	Reprocessing loss	Fuel in reactors	Temporary storage	Subtotal
U	2420000	27760	2447760	2895	18450		21345	2909	18430		21339
Pu	27490	315	27805	225	1572	929	2726	225	1560	850	2635
MA	1385	16	1401	789	4		793	4	28	135	167
FP	77910	894	78804	79140	494		79634	79570	494		80064
Total	2526785	28985	**2555770**	83049	20521	929	**104499**	82708	20512	984	**104204**

TABLE 7 The influence on high level waste (only considers actinides and fission products) generation caused by recovery ratio

Unit/t	Recovery ratio of 99.9%				Recovery ratio of 99%			
	Reprocessing loss	Fuel in reactors	Temporary storage	Subtotal	Reprocessing loss	Fuel in reactors	Temporary storage	Subtotal
U	2909	18430		21339	28880	18432		47312
Pu	225	1560	850	2635	2231	1560	872	4662
MA	4	28	135	167	41	28	134	203
FP	79570	494		80064	79005	494		79499
Total	82708	20512	984	**104204**	110156	20514	1006	**131675**

3) Under the same recovery ratio, TRU recycling can greatly reduce the amount of MA in the waste, which will bring a benefit of about 2 repositories like Yucca mountain and is suggested to give priority for development in the future.
4) The recovery ratio has significant influence on final waste characteristics, increasing the recovery ratio from 99% to 99.9% will also bring a benefit of about 2 repositories like Yucca mountain. Therefore, it is suggested that the recovery ratio in closed fuel cycle is better to be greater than 99.9%.

TABLE 8 Actinides waste characteristics after stored for 1000 years

	Radioactivity/Ci	Heat release/W	Ingestion toxicity /m^3 H2O
PWR with once through fuel cycle	5.160E+09	1.649E+08	1.167E+15
SFR with Pu recycle and 99.9% recovery ratio	9.265E+08	2.932E+07	2.144E+14
SFR with TRU recycle and 99.9% recovery ratio	6.720E+08	2.175E+07	1.577E+14
SFR with TRU recycle and 99% recovery ratio	9.310E+08	2.991E+07	2.128E+14
Yucca mountain repository under full-load	1.414E+08	4.518E+06	3.197E+13

TABLE 9 Actinides waste characteristics after stored for 1000 years (relative value)

	Radioactivity	Heat release	Ingestion toxicity
PWR with once through fuel cycle	36.5	36.5	36.5
SFR with Pu recycle and 99.9% recovery ratio	6.6	6.5	6.7
SFR with TRU recycle and 99.9% recovery ratio	4.8	4.8	4.9
SFR with TRU recycle and 99% recovery ratio	6.6	6.6	6.7
Yucca mountain repository under full-load	1	1	1

5. Conclusions

The main conclusions of this study are summarized as the followings:

1) When multi-recycled in CFR1000, the composition of Pu and integrated TRU will reach equilibrium after several recycles. For Pu equilibrium composition, the main component is 239Pu and 240Pu (70% and 25% respectively), and the fissile Pu isotope share is about 73%. For integrated TRU recycling, the MA percentage will decrease gradually, and in equilibrium state the percentage is about 3% with another 97% of Pu, and the Pu composition is similar to that of only Pu recycling. The out-of-core residence time has little influence on equilibrium composition.

2) The installed capacity in nuclear energy system with SFR matched with PWR and closed fuel cycle is very sensitive to SFR fuel out-of-core residence time and recovery ratio, while for PWR the sensitivity is really small. Reducing SFR fuel out-of-core residence time and increasing SFR recovery ratio can greatly accelerate the growth rate of the installed SFR capacity, but for the recovery ratio, it has smaller influence after more than 99%.

3) In the scenario of SFR is developed and matched with the development of PWR, there are two stages of SFR startup and SFR developing in large scale. In SFR startup stage, the changing of PWR installed capacity has great influence on the system total installed capacity, and to ensure the total system installed capacity does not decrease during the transition from PWR to SFR, the request of the fuel out-of-core residence time no longer than 5a and the recovery ratio no lower than 99% is proposed.

4) Compared to PWR with once through fuel cycle, developing SFR and closed fuel cycle can greatly reduce actinides waste generation, especially Pu, which is 10% of that in PWR with once through fuel cycle. SFR can effectively use industrial Pu, and can flexibly change the core breeding ratio to meet the requirement of keeping the total Pu amount stable in the system.

5) The integrated TRU recycling can effectively reduce and maintain the MA amount in the system, which is about 20% of that when Pu recycling. PWR with once through fuel cycle generates the most amount of MA, about 1.8 times of that in Pu recycling in SFR.

6) From the viewpoint of long term radiotoxicity of high level waste, Pu is the main source, followed by MA. Therefore, the control of Pu generation is the key to reduce long term radiotoxicity of high level waste. Compared to PWR with once through fuel cycle, the development of SFR and closed fuel cycle can reduce the long term radiotoxicity to about 18%, and the development of integrated TRU recycling can further reduce to 13%.

7) The decreasing of recovery ratio could increasing the reprocessing loss, therefore, from viewpoint of reducing waste long term radiotoxicity, the recovery ratio should achieve 99.9%. But after larger than 99.9%, the further increasing of recovery ratio will bring little benefit.

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