National Nuclear Laboratory

NWTRB Spent Fuel Management Workshop draft scenarios - results from ORION v3.12

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A report prepared for and on behalf of
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National Nuclear Laboratory

NWTRB Spent Fuel Management Workshop draft scenarios - results from ORION v3.12

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EXECUTIVE SUMMARY

This document summarises the UK National Nuclear Laboratory's (NNL) results from various fuel cycle models, the specifications of which were developed by the NWTRB as part of a benchmark exercise for the NWTRB Spent Fuel Management Workshop in June 2011.

The NWTRB requested 5 scenarios to be performed. Each scenario was a specific nuclear fuel cycle and was modelled using NNL's fuel cycle modelling code ORION. In addition to the 5 scenarios defined, a further 7 perturbation runs were also run as defined in the benchmark specification from NWTRB [1].

The first scenario (referred to as scenario 1.1 in this report) involved calculating the historic USA spent fuel inventory as of 2010. The second scenario (1.2) involved calculating the future spent fuel inventories in the USA as of 2100 from both the current reactor fleet and a future PWR and BWR fleet.

The 3rd scenario (1.3) assumed a mass of spent fuel was sent to a repository each year from 2040 onwards whereas the 5th scenario (1.5) assumed both reprocessing and disposal in addition to MOX and ERU fabrication from separated uranium and plutonium.

Within this report is a commentary discussing the results from ORION and simplifications and assumptions that were needed when developing the models in ORION.

This benchmark has highlighted several potential improvements that could be made to both ORION and other fuel cycle modelling codes currently under development. Improvements include the explicit ability to preferentially process the newest or oldest material in a buffer first and the option of choosing to define the throughput of a process plant either in terms of heavy metal mass or absolute mass.

VERIFICATION STATEMENT

This document has been verified and is fit for purpose. An auditable record has been made of the verification process. The scope of the verification was to confirm that: -

- The document meets the requirements as defined in the task specification/scope statement
- The constraints are valid
- The assumptions are reasonable
- The document demonstrates that the project is using the latest company approved data
- The document is internally self consistent

HISTORY SHEET

Issue Number	Date	Comments
Issue 1	25th May 2011	Issued to customer and to NWTRB in order to provide explanatory notes to the results from ORION. Results will be reviewed as part of benchmark and might possibly be revised following the meeting.

CONTENTS

LIST OF TABLES

Page

LIST OF FIGURES

Page

1. Introduction to ORION

ORION (current version 3.12) is a fuel cycle modelling program developed by the UK National Nuclear Laboratory (NNL). It is a Windows based program (see Figure 1) which can track up to approximately 2500 nuclides as nuclear material is moved through the fuel cycle. The list of nuclides followed can be chosen by the user. Depending on the scenario, either 2552 or 104 nuclides were tracked depending on the complexity of the scenario and the runtime.

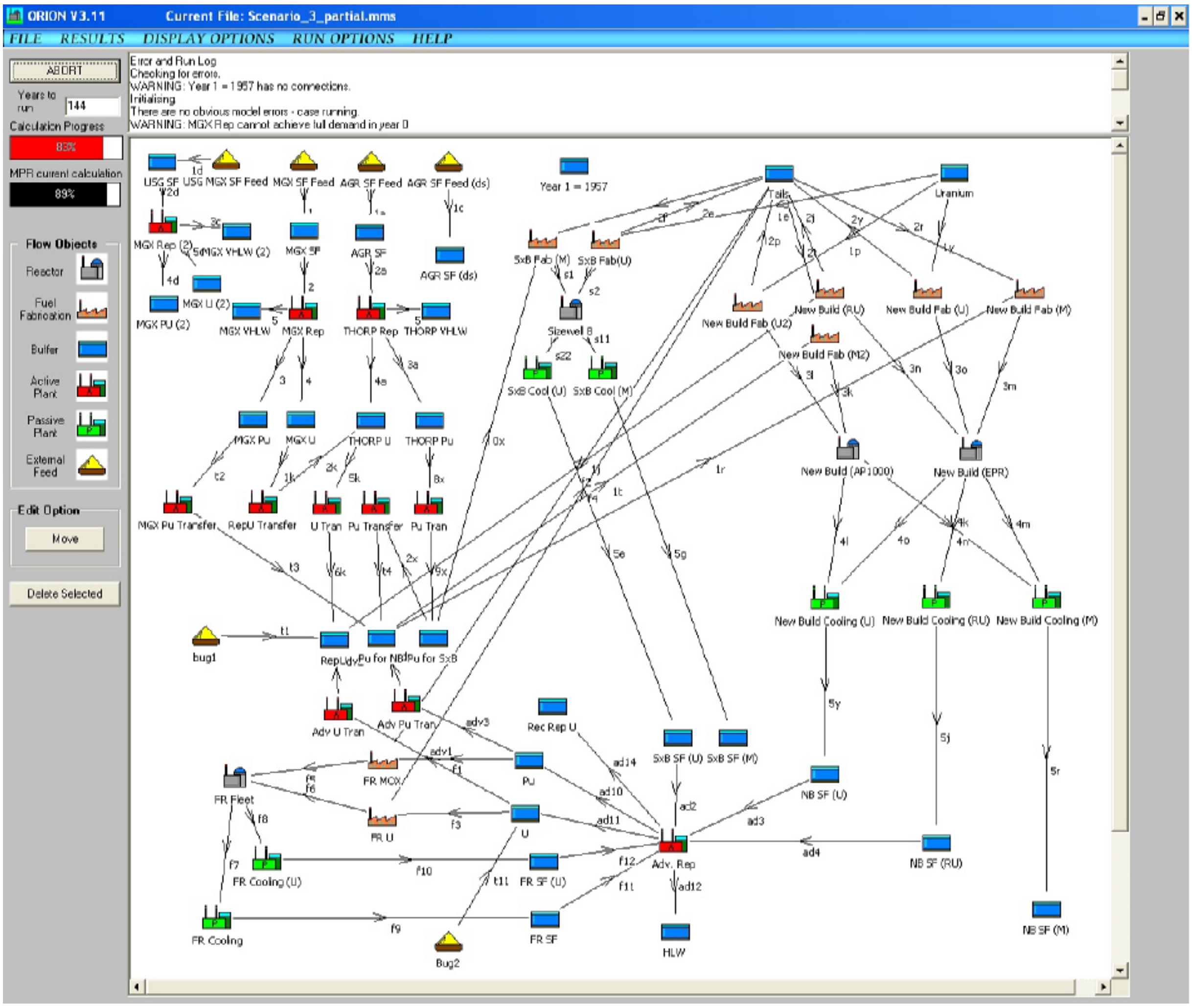
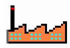


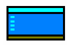
Figure 1 - Screen capture of the ORION computer program

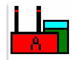
The objective of any fuel cycle code is to holistically model a fuel cycle in a <u>reasonable period of time</u>. Obviously a fuel cycle modelling program can be as complicated as possible. However, the run time for the scenario will increase. In order to ensure a fuel cycle model will run in a reasonable period of time, the smallest timestep that can be defined in ORION is 1 year. Therefore, parameters such as the irradiation time for reactor fuel and reprocessing lead times must be equal to an integer number of years. Although this might appear restrictive, such a level of detail is not usually required for a fuel cycle model.

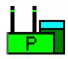
There are 6 objects that can be inserted into a fuel cycle model and are shown in Figure 1 above as an example. Since ORION is 'graphically run', it is fairly simple to develop relatively complex fuel cycle models. The 6 ORION objects are:

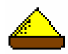
Reactor: 

Fuel Fabrication plant: 

Buffer: 

Active Plant: 

Passive Plant: 

External Feed: 

ORION objects can be dragged into the model and linked together as shown in Figure 1. Once all of the objects have been linked and defined, the fuel cycle duration is chosen and the case is run.

A brief description of each ORION object is provided in Appendix A.

2. Description of NWTRB Benchmarks

The NWTRB benchmark specification [1] defined 5 fuel cycle scenarios of increasing complexity. For some scenarios, additional perturbation cases were needed that looked at, for example, the impact of a higher reprocessing throughput on the fuel cycle.

Draft scenarios were sent by NWTRB on the 4th April [1]. This document has been used when developing the fuel cycle models. Since writing this report, a slightly modified version of the draft scenario descriptions document was obtained (which has been attached as an Appendix to this report). However, the changes are slight and in most parts confirmed assumptions that were made already.

Below is a description of each scenario. Table 1 summarises the differences in a more convenient form.

2.1. Scenario 1.1 (referred to as 2.1 in latest draft scenario descriptions document)

The purpose of this scenario was to calculate the inventory of fuel from the current PWR and BWR fleet. A reference supplied by NWTRB was used as a basis for estimating the total mass of fuel at each of the nuclear reactor parks in the USA (fuel masses in wet and dry storage were given). This reference estimated there to be approximately 62000tHM of spent fuel in wet and dry storage as of December 2009. Assuming a fixed burnup and initial ^{235}U enrichment for all PWR and BWR spent fuel (39 GWd/tHM for PWR and 32 GWd/tHM for BWR fuel), the combined spent fuel inventories were estimated using the tools available (CASMO-4 to generate cross section libraries and FISPIN to perform fuel inventory calculations).

2.2. Scenario 1.2 (2.2)

This scenario follows from scenario 1.1, except future spent fuel compositions are also calculated. The current reactor fleet is allowed to operate until the end of their quoted shut down dates with an average discharge burnup of 55 GWd/tHM. A new build fleet also comes on line in such a way as to maintain the 100.3 GWy(e) nuclear generation capacity of the USA (i.e. the first new build reactor comes on line in 2012 due to Vermont Yankee NNP shutting down). Obviously this is not realistic, but for the purposes of a benchmark is not an issue.

2.3. Scenario 1.3 (2.3)

This scenario follows from scenario 1.2, however unlike the previous scenario, fuel is sent to a repository at a constant yearly rate from 2040 onwards. Two perturbation cases were run. The first assumed a repository receipt rate of 1500 MT/year, whereas the second perturbation case assumed a repository receipt rate of 3000 MT/year. Spent fuel destined for the repository had to be at least 10 years old and the oldest fuel was to be processed first.

2.4. Scenario 1.4 (2.4)

This scenario followed from scenario 1.3. However, rather than disposal of fuel and relying on a finite resource of spent fuel from the current reactor fleet, an unlimited (and un-aging) source of spent fuel was reprocessed at a constant rate. The separated uranium and plutonium product from reprocessing was then used to fabricate MOX and enriched reprocessed uranium (ERU) fuel for a PWR reactor fleet. Although the scenario description does not explicitly state this, it was assumed the same new build fleet as given for scenarios 1.2 and 1.3 (i.e. a combined PWR and BWR fleet totalling 100.3 GWy(e)) was to be modelled. Unlike other scenarios, reactor operations were assumed to be at 'steady state' so nuclear power plants were assumed to run indefinitely (i.e. no start up or shutdowns).

Since the composition of fuel being reprocessed does not vary over the time, the composition of fuel in the separated plutonium and uranium buffers, the fresh fuel compositions and therefore the spent fuel compositions will also remain constant.

6 perturbation cases were performed:

- Scenario 1.4/1: reprocessing rate 1500MT/yr; fuel prior to reprocessing 5 years old
- Scenario 1.4/2: reprocessing rate 1500MT/yr; fuel prior to reprocessing 25 years old
- Scenario 1.4/3: reprocessing rate 1500MT/yr; fuel prior to reprocessing 50 years old
- Scenario 1.4/4: reprocessing rate 3000MT/yr; fuel prior to reprocessing 5 years old
- Scenario 1.4/5: reprocessing rate 3000MT/yr; fuel prior to reprocessing 25 years old
- Scenario 1.4/6: reprocessing rate 3000MT/yr; fuel prior to reprocessing 50 years old

2.5. Scenario 1.5 (2.5)

This scenario followed from scenario 1.3. However, in this scenario fuel is either disposed of in a repository or reprocessed. As was the case with scenario 1.3, the oldest fuel is disposed of first in a repository. Conversely, the newest fuel (fuel must be at least 5 years aged) is reprocessed first. In addition only PWR fuel (standard UO_2) is reprocessed. BWR, spent MOX and spent ERU fuel is not reprocessed and will eventually be disposed of in the repository.

Table 1 summarises the differences between the 5 different scenarios and 7 additional perturbation runs.

Table 1 - Summary of NWTRB scenarios

Scenario Property	Scenario											
	1.1	*1.2*	*1.3/1*	*1.3/2*	*1.4/1*	*1.4/2*	*1.4/3*	*1.4/4*	*1.4/5*	*1.4/6*	*1.5/1*	*1.5/2*
Calculate spent fuel composition before 2010 from current reactor fleet?	✓	✓	✓	✓	✗	✗	✗	✗	✗	✗	✓	✓
Calculate spent fuel composition after 2010 from current reactor fleet?	✗	✓	✓	✓	✗	✗	✗	✗	✗	✗	✓	✓
Dispose of fuel in repository?	✗	✗	✓	✓	✗	✗	✗	✗	✗	✗	✓	✓
Reprocess PWR fuel?	✗	✗	✗	✗	✓	✓	✓	✓	✓	✓	✓	✓
Fabricate MOX and ERU?	✗	✗	✗	✗	✓	✓	✓	✓	✓	✓	✓	✓
Repository throughput	n/a	n/a	1500	3000	n/a	n/a	n/a	n/a	n/a	n/a	1500	1500
Reprocessing throughput	n/a	n/a	n/a	n/a	1500	1500	1500	3000	3000	3000	1500	3000
Spent fuel reprocessed?	n/a	n/a	n/a	n/a	D1	D2	D3	D1	D2	D3	D4	D4

D1: tHM PWR 4.4w/o enriched natural uranium fuel, 55 GWd/tHM, 5 years cooled

D2: PWR 4.4w/o enriched natural uranium fuel, 55 GWd/tHM, 25 years cooled

D3: PWR 4.4w/o enriched natural uranium fuel, 55 GWd/tHM, 50 years cooled

D4: PWR UO_2 (not ERU) spent fuel (either from current reactor fleet or from new reactor fleet)

3. Limitations of ORION, impact on NWTRB scenarios and assumptions made when setting up fuel cycle models

This section explains how the NWTRB benchmarks were modelled using ORION and various modelling assumptions that were made that could potentially lead to differences with other benchmark applicants results.

3.1. Preferential processing of material

Several of the NWTRB benchmarks (1.2, 1.3, 1.4 and 1.5) state a preference over which fuel is reprocessed or disposed of first. However, the age of material (as in time since discharge) is not attributed to material by ORION since any new material entering a buffer is mixed with material already present. This makes it impossible to discriminate between ‘old’ and ‘new’ material which enters the same ‘buffer’ object (i.e. if fuel from the current reactor fleet is discharged in the year 2000, this material will be added to other fuel already discharged prior to this date).

As an approximate work around to this problem, present and future materials are divided into separate streams. With ORION it is possible to preferentially process particular input streams. Therefore, spent fuel discharged from the current reactor before and after 2010 as well as newer spent fuel from the new build fleets are preferentially treated since these materials are already segregated and present in different buffers.

3.2. Fuel Assembly Tracking

ORION works with masses and not necessarily the number of fuel assemblies. However if spent fuel is segregated by reactor type in the model (which it is) and the mass of each assembly is known and is the same throughout the scenario, the equivalent number of assemblies can be determined by dividing one by the other.

3.3. Processing Plant Throughput definition

In ORION, throughputs for processing plants (i.e. repository / reprocessing facility) are defined as heavy metal mass rather than total fuel mass. Therefore, the masses of fuel reprocessed / disposed each year will be slightly higher than 1500MT or 3000MT, the difference being the fission product mass of the fuel.

3.4. Enrichment penalty for enriched reprocessed uranium fuel

Although the NNL owns a program (RuCalc) that calculates the residual absorption penalty of ^{232}U, ^{234}U and ^{236}U in enriched reprocessed uranium (ERU) fuel, ORION does not have the capability to automatically vary the ^{235}U enrichment necessary to attain the same reactivity of standard UO2 fuel at a given enrichment It does however solve the multi isotope balance equations so the composition of ^{232}U, ^{234}U and ^{236}U in the enriched fuel will be accurate. Therefore, an indicative uranium vector was extracted from ORION and used in RuCalc to determine what the equivalent ERU enrichment would be. The same ERU enrichment was assumed throughout the ORION fuel cycle model (benchmarks 1.4 and 1.5) for all ERU fuel even though there might be slight changes in separated uranium quality over time that will impact the level of enrichment actually required.

3.5. Yearly Time Steps

The time step in ORION is 1 year. Due to simplifications in the reactor model within ORION, a fraction of the core needs to be discharged every year and the dwell time must be an integer number of years long. This simplification limits the burnups that can be modelled in ORION. Since the dwell time must be integer number of years long the burnup will be:

Burnup = Dwell Time (years) x 365.25 x Load Factor (fraction) x Power Density (W/gHM)

For example, if a given reactor has a power density of 30 W/g and the load factor is 90%, the burnup modelled can only be changed by varying the dwell time. Since the dwell time can only be an integer number of years, the burnups possible will be 9.8, 19.7, 29.6, 39.5, 49.3 etc GWd/tHM.

In order to choose any burnup, the core mass is varied until the power density is such that the burnup is achieved in a given number of years. Varying the input parameters in such way results in the correct yearly mass discharges and burnups. However, the final discharge mass at reactor shutdown will be slightly incorrect depending on how much the core mass is varied to attain the desired burnup.

3.6. Simplifications to ORION fuel cycle models

In order to simplify the ORION model (and to reduce calculation time; especially for benchmark 1.5) individual reactors were not modelled. Instead reactors were grouped together to form a reactor 'unit'. Each reactor unit has a combined electrical power rating equal to a number of reactors which it is intended to model. The new build fleet of 100.3 GWy(e) comprised of 22 x 3032.32 MW(e) PWR units and 11 x 3061.84 MW(e) BWR units. The benchmark specification stated the new build fleet should be such that the total nuclear generation capacity remains at 100.3 GWy(e). However, since the total generation capacity of the new build fleet will change by steps of either 3032.32 or 3061.84 MW(e), it was not possible to keep the capacity exactly at 100.3 GWy(e).

3.7. Modelling assumptions

For scenario 1.4, the size of the new build fleet as well as the average discharge burnups were not given in the specification. If it was not for the fact that one of the results required was the 'percent reduction in total natural demand' (rather than an absolute change), it is believed the results would be insensitive to the new build fleet size assumed. For this scenario, the same new build fleet size and average discharge burnup as used in the previous benchmarks was assumed (i.e. 100.3 GWy(e) capacity; 66711MW(e) from a PWR fleet and 33860 MW(e) from a BWR fleet and an average discharge burnup of 55GWd/tHM).

For scenario 1.5, it was not clear whether results for every year of operation or just for the final year in the scenario (2100) were required. For simplicity only results for the final year (2100) are given. If annual results are needed, the fuel cycle could be re-run during the workshop.

For scenario 1.1, it was not obvious what age to assume for the historic PWR and BWR fuel (i.e. the fuel discharged up to December 2009). Since one of the output measures was the inventory of the spent fuel, the age of the fuel would be a critical input parameter and will affect the final result. In the ORION model, the 40592tHM and 21104tHM of historic PWR and BWR fuel was assumed to have been discharged gradually over the years depending on which reactor units were operating. Start up and where applicable shut down dates given by NWTRB were used to determine how much fuel was discharged each year. Overall the total mass of fuel discharged was approximately 62000tHM.

4. ORION Results

Fuel cycle scenarios 1.1, 1.2, 1.3 and 1.5 are similar but each benchmark is slightly different (e.g. reprocessing / disposal / new build fleet might be disabled). Figure 2 is a screen capture for scenario 1.5. Note that these 4 scenarios track 2552 nuclides since the neutronics methods needed for these scenarios are simple and quick (i.e. the T-value method as opposed to the 'MPR' method was needed – see Appendix A for a detailed description for these methods). Scenario 1.5 however, required the use of the MPR method to calculate the spent fuel composition of MOX fuel from the PWR fleet. Since the MPR method is computationally intensive, the number of nuclides tracked was reduced to 104 (most of which are heavy metal nuclides). As such the fission product masses given in scenario 1.5 will be underestimated.

All results are contained in an Excel spreadsheet that accompanies this report. The Excel spreadsheet filename is "NWTRB_Results.xls".

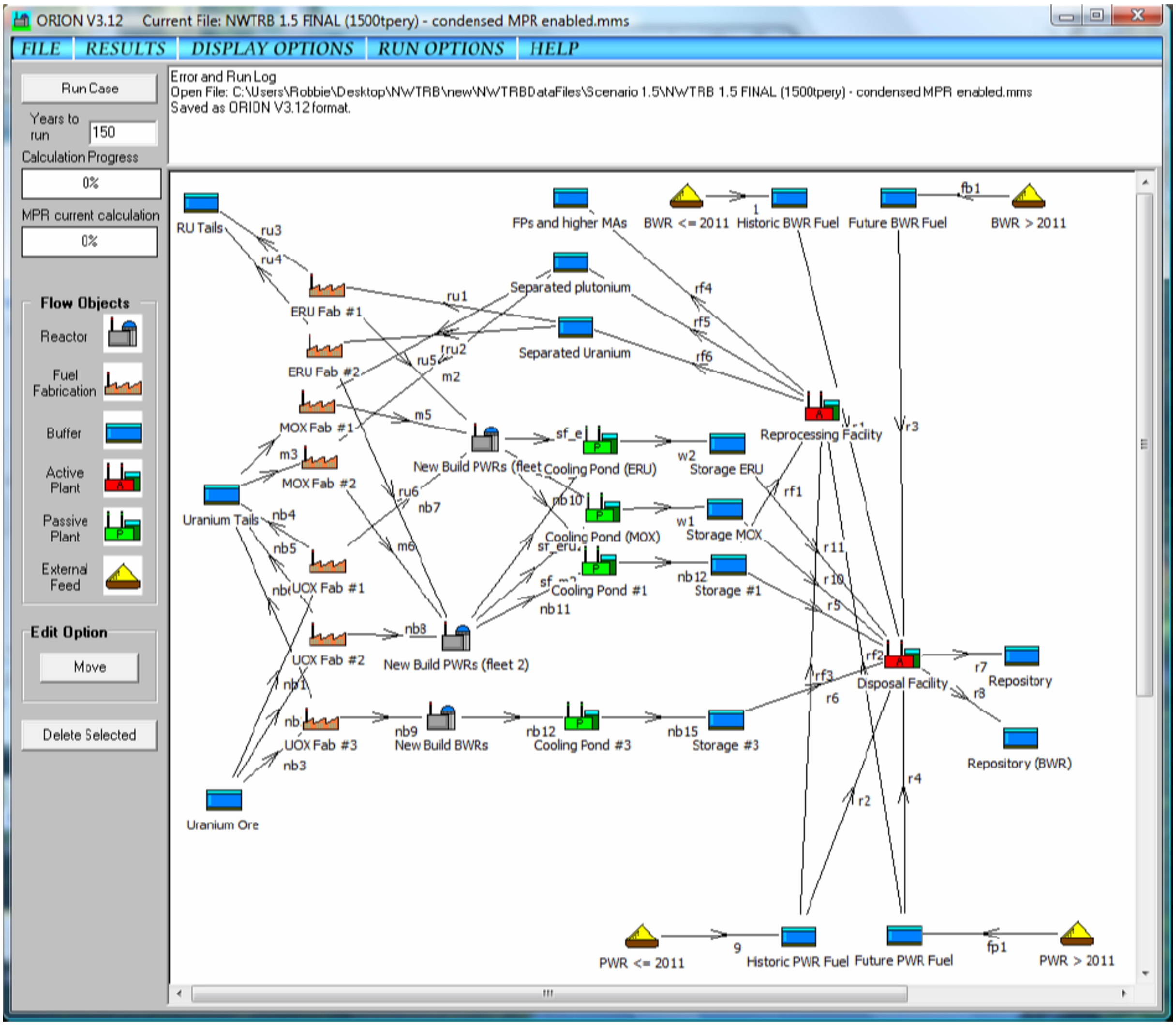


Figure 2 - scenario 1.5 in ORION

4.1. Scenario 1.1

This scenario involved calculating the composition of approximately 62000tHM of spent fuel discharged from the PWR and BWR fleets in the USA. In order to calculate the composition, FISPIN calculations were initially performed to calculate the spent fuel composition for 1 tHM of PWR and 1tHM of BWR spent fuel taken to a burnup of 39 GWd/tHM and 32 GWd/tHM respectively. The amount of material discharged each year was then calculated and the correct inventories 'injected' directly into the ORION fuel cycle model using a 'feed object' – see Figure 2 (example: object "PWR <= 2011"). The results given in the Excel spreadsheet are for the year 2010 and are present on Tab "1.1". Detailed inventories are given for the PWR and BWR spent fuel masses separately (see columns G:H and J:K). A summary of the combined inventories (cells M11:P38) is also included and gives the output measures as specified by the benchmark.

4.2. Scenario 1.2

This scenario is similar to scenario 1.1, however as well as calculating the inventory for approximately 62000tHM of historic spent fuel, the scenario also calculates the inventory of the fuel that will be discharged from the fleet over its remaining lifetime. The shutdown dates as given in the NWTRB references have been used to calculate the mass of fuel discharged from the current reactor fleet. A new build fleet has also been incorporated into the model. The size and start-up dates for each of the units in the new build fleet have been set such that the combined nuclear generation capacity remains at approximately 100.3 GWy(e). Therefore, the first unit will come on line in 2012 to maintain the same overall nuclear generation capacity as before due to the closure of Vermont Yankee in the previous year[1]. The new build reactor lifetimes were set to 60 years long, therefore in order to maintain the same generation capacity, a second future generation of reactors would come on line beginning 2072. In 2100, the generation capacity will still be 100.3 GWy(e).

The assumed burnup of the fuel from the new build fleet was 55 GWd/tHM. Due to limitations as discussed in the previous section, the core mass was modified such that the total electrical output, average discharge burnup and the total mass of fuel discharged per year is correct.

The results from this scenario are given in tab '1.2'. Full inventories in the year 2100 are given in columns G to W for the following spent fuel buffers:

- "Current BWR fleet (> 2010)" – fuel discharged after 2010 from the current BWR fleet (i.e. Susquehanna, Quad Cities, etc).
- "Current PWR fleet (> 2010)" – fuel discharged after 2010 from the current PWR fleet (i.e. South Texas, Vogtle, etc).
- "Current BWR fleet (<2010)" – fuel discharged before 2010 from the current BWR fleet. The total fuel mass should be roughly the same as given in benchmark 1.1 (approximately 21000tHM). However the inventory will have decayed for an additional 90 years.

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[1] Note that due to simplifications necessary when developing the fuel cycle, the new build PWR and BWR 'unit' sizes were set to 3030.32 and 3061.36 MW(e) respectively. Each replacement unit could in principle have been modelled exactly. However, this would have made the model overly complicated and difficult to design. Therefore the first PWR and BWR unit in the ORION model did not come on line until 2014 (further units came offline in 2012 and 2013 including Prairie Island 1 and Pilgrim).

- "Current PWR fleet (<2010)" – fuel discharged before 2010 from the current PWR fleet. The total fuel mass should be roughly the same as given in benchmark 1.1 (approximately 40600 tHM).
- "New PWR Build Fleet" – fuel discharged from the new build PWR fleet. The start-up dates for the individual units have been set such that the total nuclear generation capacity of the PWR fleet remains approximately the same.
- "New BWR Build Fleet" – fuel discharged from the new build BWR fleet. The start-up dates for the individual units have been set such that the total nuclear generation capacity of the BWR fleet remains approximately the same.

A summary table has been given (see cells Y7 to AH23) which gives the output measures defined in the benchmark specification.

4.3. Scenario 1.3

Scenario 1.3 comprises two separate benchmarks. Scenario 1.3/1 assumed a repository throughput of 1500MT/year whereas scenario 1.3/2 assumed a repository throughput of 3000MT/year.

In ORION it is only possible to define heavy metal mass throughputs for processing plants. The actual fuel masses will be slightly higher due to the additional fission product mass.

Also, as described in the previous section, it is not possible to preferentially choose to process either the newest or oldest material in a buffer since any material added to a buffer is mixed with any material already present. This benchmark has however highlighted a potential improvement which could be made to ORION and other more advanced fuel cycle modelling tools under development. Spent fuel is however contained in 6 separate storage buffers:

- PWR fuel discharged before 2010 (current fleet)
- PWR fuel discharged after 2010 (current fleet)
- BWR fuel discharged before 2010 (current fleet)
- BWR fuel discharged after 2010 (current fleet)
- All PWR fuel discharged from new build fleet
- All BWR fuel discharged from new build fleet

Since the benchmark stated the oldest fuel should be disposed of first, fuel discharged before 2010 was preferentially treated before fuel discharged after 2010 and fuel from the new build fleet.

A summary of PWR and BWR fuel disposed per year is given in tab '1.3_1' (1500MT/year) and '1.3_2' (3000MT/year) (see cells H6 through to L158).

4.4. Scenario 1.4

Scenario 1.4 comprised 6 similar benchmarks. Figure 3 is a screen capture of the scenario. Each 'sub-benchmark' assumed either a different reprocessing capacity or a different spent fuel age prior to reprocessing (i.e. a different post irradiation storage time (PIST)).

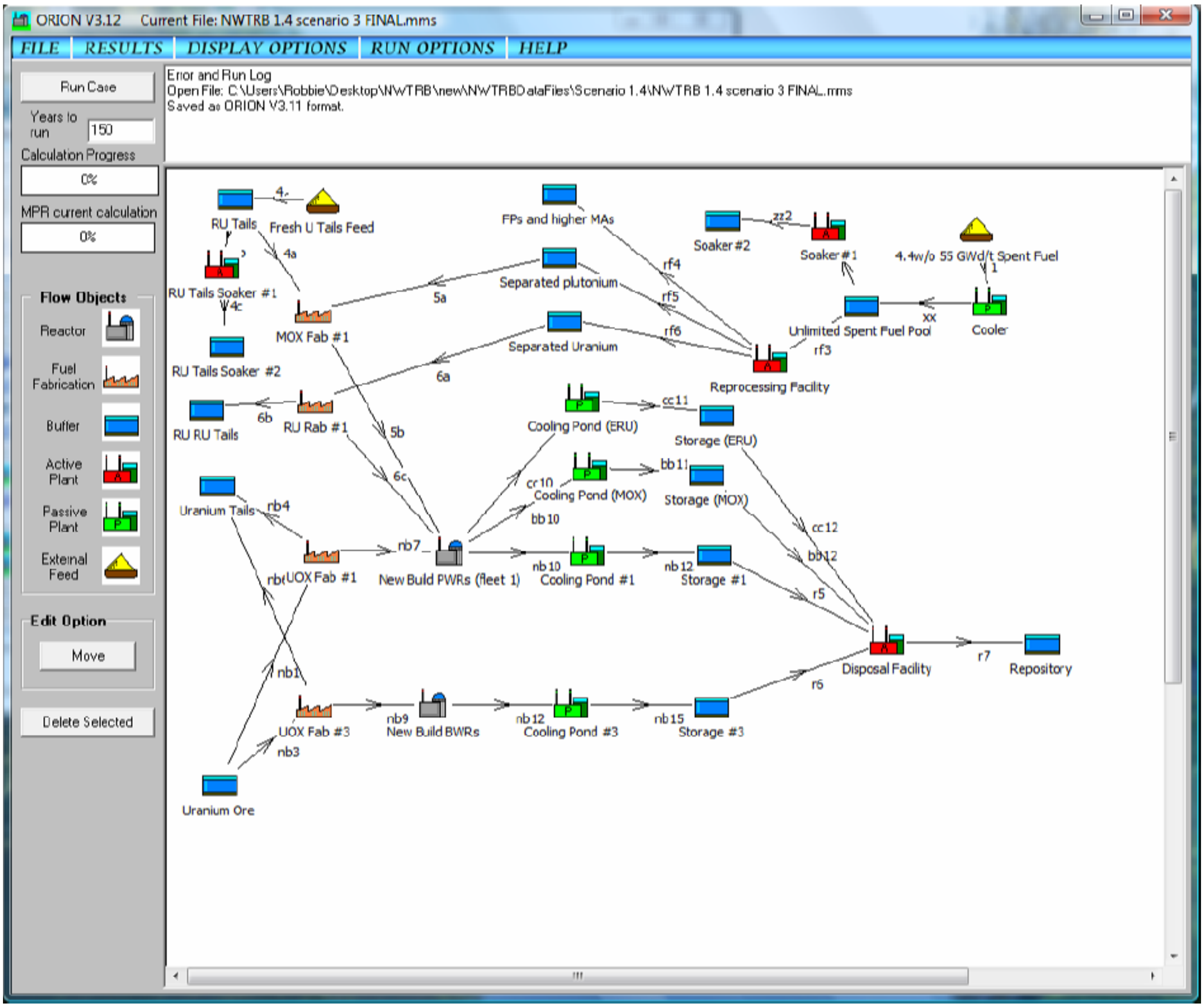

Figure 3 - Scenario 1.4 in ORION

The benchmark specification did not give any details regarding the reactor park for this scenario. However, the vast majority of the results requested will be fairly insensitive to what is assumed. For simplicity, the same new build fleet as used in scenario 1.3 has been assumed (i.e. 100.3 GWy(e) capacity; 66711MW(e) from a PWR fleet and 33860 MW(e) from a BWR fleet and an average fuel discharge burnup of 55GWd/tHM).

These scenarios have been set up in such a way that the fuel composition prior to reprocessing is constant over time and will always be either 5, 25 or 50 years old. Exactly 1500MT or 3000MT of spent fuel (heavy metal) is reprocessed each year. The separated plutonium and uranium is then used to fabricate either MOX fuel (in conjunction with an unlimited supply of tails) or enriched reprocessed uranium fuel (ERU). There is no limit on MOX or ERU use in the PWR fleet, but the amount of fuel loaded will depend on what material is available.

All scenarios will reach equilibrium after about 55 years and will stay at equilibrium until the scenario ends. Results for each of the 6 scenarios are given on 6 separate tabs:

- Scenario 1 (reprocessing capacity: 1500MT/year fuel age: 5 years) – tab '1.4_1'
- Scenario 2 (reprocessing capacity: 1500MT/year fuel age: 25 years) – tab '1.4_2'
- Scenario 3 (reprocessing capacity: 1500MT/year fuel age: 50 years) – tab '1.4_3'
- Scenario 4 (reprocessing capacity: 3000MT/year fuel age: 5 years) – tab '1.4_4'
- Scenario 5 (reprocessing capacity: 3000MT/year fuel age: 25 years) – tab '1.4_5'
- Scenario 6 (reprocessing capacity: 3000MT/year fuel age: 50 years) – tab '1.4_6'

The format of the scenario results on each tab is similar. A table is given from cell G5 through to N47. Yearly inventories for each of the 4 fresh fuel streams (PWR MOX, PWR ERU, standard PWR UO_2 and standard BWR UO_2 fuel) are given for the main isotopes (uranium, plutonium and Am241). All other minor actinides are given as 'other'.

Also given is the inventory of MAs and FPs from reprocessing per year as well as uranium ore yearly requirements and the mass of tails generated each year. The mass of uranium tails from enriching uranium ore and RepU per year have been calculated. The reduction in uranium ore requirements (relative to the case where only standard UO_2 fuel is loaded) has been given as an absolute figure and as a percentage.

4.5. Scenario 1.5

Scenario 1.5 involves a reprocessing and fuel disposal strategy. Since the fresh fuel composition for MOX (and ERU to an extent) will vary over time and the benchmark specification requires these compositions to be calculated (unlike scenario 1.4), the 'MPR' method was used in ORION to calculate all PWR spent fuel inventories (see Appendix A). The 'MPR' method involves calculating a cross section library previously using a neutronics code such as CASMO which is then used by ORION to calculate the spent fuel inventory directly. Note that for other scenarios, a much simpler approach whereby the inventories were entered manually for each reactor object was used (T-value method).

Since the 'MPR' method was required to calculate MOX spent fuel inventories, the number of nuclides had to be reduced from 2552 to 104 nuclides. If 2552 nuclides were tracked by ORION, the calculation time would have been a prohibitively long. Tracking 104 nuclides still took a long time (3 – 4 hours) to perform the calculation but was manageable.

The benchmark specification contained two scenarios with different reprocessing throughputs, 1500MT/year and 3000MT/year. The Pu content in MOX fuel necessary to have the same equivalent reactivity as 4.4w/o standard UO_2 fuel was calculated previously assuming a reference Pu vector. Effective fissile coefficients were also calculated using in-house tools and used by ORION to automatically determine the Pu content needed depending on the separated Pu quality in any one year. Note that although the scenario stated the plutonium content could not exceed 14w/o, the actual Pu content needed by the PWR fleet never reached this limit. Furthermore it is not currently possible to set a Pu content limit in ORION so it is fortuitous the limit was never breached.

As with other scenarios, it is not possible to preferentially process the newest or oldest fuel. However PWR and BWR fuel is segregated in the model so can be treated separately. Also the spent fuel streams (fuel discharged before and after 2010 from the current reactor fleet and fuel discharged from the new build fleet) are preferentially treated by the reprocessing facility and repository.

The tails masses given in the results do not include any tails production from enriching UO_2 fuel for the current reactor fleet. This is due to the method used to incorporate the spent fuel compositions into ORION for the current reactor fleet. The tails masses only include tails from fabricating fuel for the new build fleets. Similarly, the uranium ore requirements for the current reactor fleet are not modelled due to the method used to setup the current reactor fleet in the ORION model. Only uranium ore requirements for the new build fleet are accounted for (i.e. reactor objects "New Build PWRs (fleet 1)" and "New Build BWRs" as shown in Figure 3).

The results for the two 1.5 scenarios are given on separate tabs:

- Scenario 1: 1500MT/year reprocessing throughput: tab '1.5_1'

- Scenario 2: 3000MT/year reprocessing throughput: tab '1.5_2'

On each tab, the repository inventory for HLW / MAs from reprocessing is given (see column H). Note that separation factors of 0.9999 have been assumed for Pu and U separation (i.e. 0.01% of all Pu and U from reprocessing spent fuel is sent to the waste stream along with all MAs and fission products). The inventories have been calculated for 2100. The total mass of PWR and BWR fuel are given in the table between cells L7:O12. The compositions of fresh fuel fabricated in the final year of the scenario (2100) has been calculated (i.e. the fuel fabricated in the final year only) and are given in the table between cells L16:N44.

In order to calculate the percent reduction in total natural uranium demand, the total uranium ore used by the new reactor fleet by 2100 was compared against the same value from benchmark 1.2 (i.e. identical benchmark except with no re-use of material). Note that because the uranium ore requirements for the current reactor fleet are not modelled (i.e. uranium ore is only used to fabricate fresh fuel for the new build fleets), the percentage reduction is relative to the uranium ore requirements for the new build fleet only.

5. Conclusions

This report summarises work carried out by the NNL to perform fuel cycle modelling assessments for various fuel cycle scenarios defined by the NWTRB as part of a benchmark exercise. The work has highlighted some potential improvements that could be made to ORION and other more advanced tools currently in development. Potential improvements include the ability to preferentially process the newest or oldest material in a buffer first and the option of choosing to define the throughput of a process plant either in terms of heavy metal mass or absolute mass. At the moment, ORION can preferentially process material from a given stream but it can’t preferentially choose to process either the newest or oldest material from a particular stream. When defining processing plants in ORION, the throughputs are always given in terms of heavy metal mass. It might be useful to allow the user to choose total material masses.

6. References

1 "NWTRB Spent Fuel Management Workshop Draft Scenarios", 4th April 2011 (since this an internal memo, the document has been appended to this report as Appendix B)

Appendix A – ORION Object Descriptions

Introduction to the Reactor Object

When setting up a reactor in ORION, parameters such as uranium enrichment, cycle length, core mass, dwell time, load factor and power density need to be defined by the user. However there are certain restrictions on what can be defined:

- The fuel enrichment / fissile fraction of the fuel can be defined by the user but in the current version can not be varied over time[2];
- The fuel loading strategy cannot vary over time;
- Since ORION timesteps are always 1 year, the dwell time must be equal to an integer number of years;
- A fuel reload must occur at the start of each year whilst the reactor is operational – this effectively means the cycle length must be 12 months long.

A reactor can have more than 1 fuel type defined and can preferentially draw material from a particular stream. In this study, all reactors using MOX were defined in such way as to preferentially use MOX fuel. If there is insufficient MOX fuel, the shortfall is made up using additional uranium fuel. Different cross section libraries can be defined for each fuel type so the inventory calculations are sufficiently accurate. All cross section libraries are either generated using the reactor physics codes CASMO-4 or ERANOS for thermal and fast reactor systems respectively. In addition the different fuel types can be treated individually and sent to different cooling ponds for processing.

The reactor object transmutes material using one of two calculation methods:

- "T-Value" method
- "MPR" method

A brief explanation of the two methods used in ORION to calculate the spent fuel inventory from a reactor object follows:

"T-Value Method"

If the "T-value method" is used to model the transmutation of fuel in a reactor, a neutronics code such as WIMS or CASMO-4 is first used to calculate a cross section library. This library is then used in an inventory code such as FISPIN or ORIGEN to calculate the spent fuel inventory[3]. Next, the spent fuel inventory is then exported to ORION and used directly to simply transform the fresh fuel feed to a spent fuel composition. This method is relatively quick and should only be used for reactors where the fresh fuel composition does not vary much over time (e.g. reactors using UO_2 fuel only). For reactors which are MOX fuelled for example, the input feed will vary

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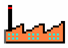
[2] Note that a set of Pu equivalence factors can be used to automatically calculate the required fissile fraction for equivalence with a reference case.

[3] The spent fuel inventory could be extracted directly from a neutronics code such as WIMS or CASMO-4 however since ORION tracks far more nuclides than traditional lattice codes, some nuclides important from a decay heat or radiotoxic viewpoint (but neutronically inert) will be missing from the fuel cycle model.

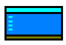
significantly over time and a more sophisticated approach (the MPR method) is used to calculate the spent fuel inventory. In this study the "MPR method" is used in scenario 1.5 only.

"MPR Method"

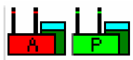
The MPR method uses a condensed 1-group cross section library to calculate the spent fuel inventory for a reactor object. A neutronics program such CASMO-4 or WIMS is used to generate burnup dependent cross section data which is then condensed down to 1 group. The method is very generic and any nuclear reaction can be included in the cross section library. However for this study only (n,γ), (n,2n) and (n,fission) reactions were considered. In addition a set of Pu equivalence factors can be defined and used to automatically calculate the fissile mass required to maintain criticality. It is especially important to model MOX fuel irradiations using the MPR approach in scenarios where multiple recycle of actinides results in a fissile feed vector that continually evolves over time. The fissile feed vector will of course directly affect the spent fuel composition – it would not be possible to capture this effect using the simpler T-value approach.

Introduction to the Fuel Fabrication Plant Object 

A fuel fabrication plant object can either enrich and fabricate UO_2 fuel, or fabricate MOX fuel from a fissile feed and carrier stock. If the object is set up to fabricate UO_2 fuel, a tails enrichment has to be defined. For this series of benchmarks a tails enrichment of 0.2 w/o has been assumed for all UO_2 fabrication plants regardless of input feed enrichment.

Introduction to the Buffer Object 

A buffer in ORION simply holds material until an object downstream (such as a reprocessing facility or fabrication plant) requires it. Note that no segregation is possible in an ORION buffer. Including such a capability in ORION would increase the run time drastically. Therefore when new material is transferred in to a buffer, it is incorporated with the material already present. When an active plant object (such as a reprocessing plant) draws material from a buffer, the material will be an average of the total content present. This negates the ability for ORION to preferentially choose to process either the newest or oldest fuel present in a buffer.

Introduction on the Active and Passive Plant Object 

An active/passive plant object is used to partition material from an input feed. In this scenario, active plants are used to simulate the separation of material in a reprocessing facility. For example, the object titled "Reprocessing Facility" in scenario 1.5 simulates the reprocessing of PWR fuel and moves 99.99% of all uranium to the "Separated Uranium" buffer, 99.99% of all plutonium to the "Separated Plutonium" buffer, and everything else to the "FPs and higher MAs" buffer (including 0.01% Pu and U). An active plant allows the user to limit the yearly throughput whereas a passive plant only allows for an unlimited throughput.

ORION output

ORION can extract and plot the masses of each nuclide flowing between objects as well as the masses contained in any buffer (). In addition, a set of nuclide metrics are defined within ORION which when multiplied by the nuclide mass allows the program to calculate the following parameters:

- Activity (Bq)
- Radiotoxicity (Sv)
- Toxic potential (m^3)
- Spontaneous neutron emission (s^{-1})
- Heat production (W)

The data is plotted on screen by ORION (see Figure 4) but can also be saved to an external file which can then be exported into Excel for further analysis.

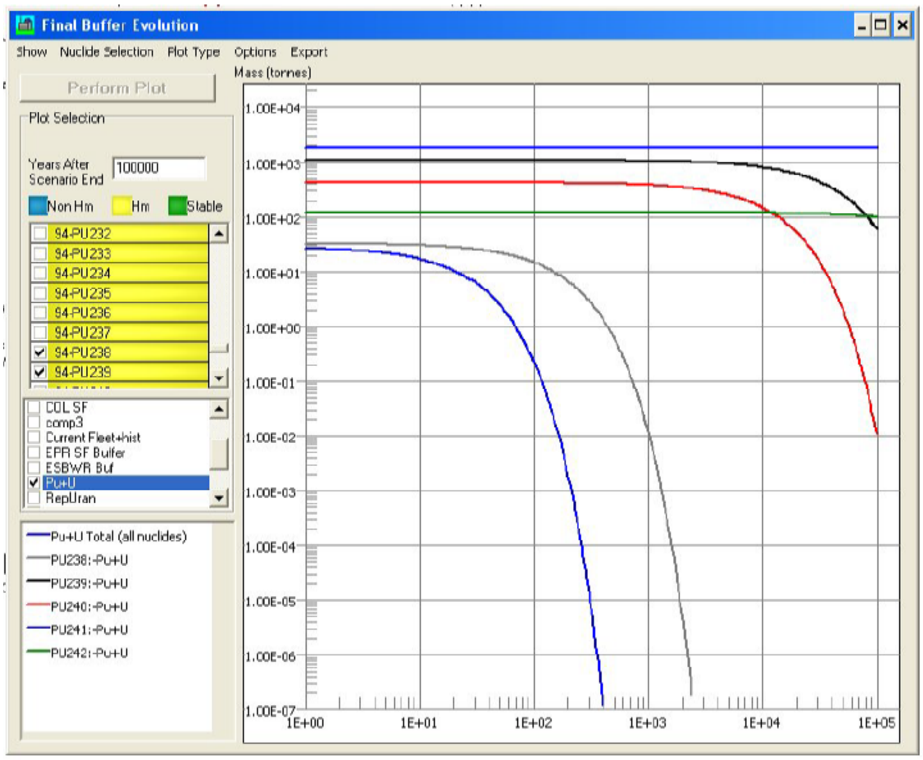


Figure 4 - Typical results from ORION. The figure shows the evolution of the main plutonium isotopes over time for a buffer

Appendix B – NWTRB Scenario Description

This is the most recent scenario descriptions document dated 12th May 2011. Note that the ORION scenarios were developed using the 4th April versions, however these are virtually identical.

NWTRB Workshop on Evaluation of Waste Streams Associated with LWR Fuel Cycle Options

Draft Scenarios

1. Purpose

The objectives of the workshop are to:

1) Establish consistency in input assumptions for the calculations of spent fuel generation and management in the U.S.
2) Understand how the scenario definitions provided by the NWTRB are applied in the calculation of spent fuel characteristics. If there are differences in the spent fuel characteristics, identify why they exist.
3) Compare analysis results, in sequence, using the scenario definitions below.

2. Analysis Phases

2.1. Characteristics of U.S. Spent Fuel Inventory as of December 2009

2.1.1. Assumptions

1) The attached file (ExistingPlantData_06March2011.pdf) provides present nuclear power plant characteristics and the wet and dry storage inventories as of December 2009. The information was obtained from:
   - For present operating nuclear plants: *U.S. Energy Information Administration* web page: *http://www.eia.doe.gov/cneaf/nuclear/page/operation/statoperation.html*
   - For spent fuel storage pools and reactor core sizes: DOE Total System Model file, *Pool Capacities_012309CB.*
   - For the number of assemblies in storage and the average characteristics of the assemblies in storage: DOE Total System Model file TSMPP_SNF_Discharge_09_052809.xls.
2) All PWR assemblies contain an initial uranium mass of 0.43 MTU, an initial ^{235}U enrichment of 3.43% and a burn-up of 39 GWd/MT.
3) All BWR assemblies contain an initial uranium mass of 0.18 MTU, an initial ^{235}U enrichment of 2.39% and a burn-up of 32 GWd/MT.

2.1.2. Output Measures

Based on the assumptions in Section 2.1.1, calculate the following:

1) Total mass of spent fuel at the beginning of 2010.
2) Total mass of ^{234}U, ^{235}U, ^{236}U, and ^{238}U in spent fuel at the beginning of 2010.
3) Total mass of ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu in spent fuel at the beginning of 2010.

4) Mass of fission products and minor actinides, either total or by isotope, in spent fuel at the beginning of 2010.

2.2. Spent Fuel Discharged Through 2100

2.2.1. Assumptions

1) The assumptions in Section 2.1.1.
2) Nuclear power plant operation starts on January 1 of the year of commercial operation and all plants operate for 60 years.
3) Sufficient new nuclear power plants will come on line to maintain the current generation capacity of 100.3 Giga-watts (electrical).
4) A plant capacity factor of 90% (100% of design thermal power for 90% of the time each year).
5) From 2010 through the end of plant life, PWR fuel assemblies discharged have an initial ^{235}U enrichment of 4.4% and a burn-up of 55 GWd/MT.
6) From 2010 through the end of plant life, BWR fuel assemblies discharged have an initial ^{235}U enrichment of 4.35% and a burn-up of 55 GWd/MT.
7) No reprocessing available before 2100.
8) No repository available before 2100.

2.2.2. Output Measures

Based on the assumptions in Section 2.2.1, calculate the following:

1) Total number of PWR assemblies discharged.
2) Total number of BWR assemblies discharged.
3) Total mass of ^{234}U, ^{235}U, ^{236}U, and ^{238}U discharged.
4) Total mass of ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu discharged.
5) Mass of fission products and minor actinides discharged, either total or by isotope.

2.3. Impact of Repository Disposal

2.3.1. Assumptions

1) The spent fuel discharged in Section 2.2.
2) No reprocessing available before 2100.
3) The repository starts operation in 2040 and begins at full capacity of:
   - Scenario 1 - 1,500 MT/year
   - Scenario 2 - 3,000 MT/year
4) Spent fuel must be at least 10 years old for repository disposal and fuel selection starts with oldest fuel first.

2.3.2. Output Measures

Based on the assumptions in Section 2.3.1, calculate the following:

1) Total mass of PWR spent fuel disposed of each year through year 2100 for each scenario.
2) Total mass of BWR spent fuel disposed of each year through year 2100 for each scenario.

2.4. Steady State Reprocessing and Fabrication of PWR MOX and Recycled UOX Fuel

2.4.1. Assumptions

1) There is a sufficient quantity of spent PWR fuel with the following characteristics for a reprocessing facility to operate at full capacity:
   - Fabricated using new uranium
   - Initial enrichment – 4.4%
   - Burn up – 55 GWd/MT
2) Only PWR fuel of this type is reprocessed
3) All other spent fuel is stored
4) PWR MOX assemblies are fabricated from separated plutonium and fresh uranium tails (^{235}U assay in tails mass is 0.2%). MOX assemblies are limited to a maximum total plutonium content of 14%. No BWR MOX assemblies are fabricated.
5) PWR recycled UOX assemblies are fabricated from enriched recycled uranium (no blending of highly enriched uranium with the separated uranium). There is no limit on the maximum ^{235}U assay in the recycled UOX assemblies to off-set the loss of reactivity because of ^{236}U content. No BWR recycled UOX assemblies are fabricated.
6) There is an unlimited amount of natural uranium, natural uranium enrichment capacity, and new uranium UOX assembly fabrication capacity.
7) All operations are at steady state:
   - Nuclear power plants - no new or replacement units starting up
   - Reprocessing facility – operating at full capacity
   - MOX fuel fabrication facility – sufficient capacity to recycle all separated plutonium
   - Recycled UOX fuel fabrication facility – sufficient capacity to recycle all re-enriched separated uranium
8) There are six scenarios:
   - Scenario 1 - Reprocessing capacity of 1,500 MT/year and all fuel 5 years old
   - Scenario 2 - Reprocessing capacity of 1,500 MT/year and all fuel 25 years old
   - Scenario 3 - Reprocessing capacity of 1,500 MT/year and all fuel 50 years old
   - Scenario 4 - Reprocessing capacity of 3,000 MT/year and all fuel 5 years old
   - Scenario 5 - Reprocessing capacity of 3,000 MT/year and all fuel 25 years old
   - Scenario 6 - Reprocessing capacity of 3,000 MT/year and all fuel 50 years old

2.4.2. Output Measures

Based on the assumptions in Section 2.4.1, calculate the annual values of the following:

1) Mass of fission products and minor actinides separated by reprocessing, either total or by isotope
2) Percent reduction in total natural uranium demand.
3) Either total number or mass, and isotopic composition, of assemblies fabricated:
   - New uranium PWR assemblies
   - New uranium BWR assemblies
   - PWR recycled UOX assemblies – all equivalent to 4.4% natural ^{235}U enrichment
   - PWR MOX assemblies (including Pu quality, Pu percent)
4) Mass of uranium tails generated:

- New uranium tails
- Recycled uranium tails

2.5. Impacts of Reprocessing Combined With Repository Disposal

2.5.1. Assumptions

1) The spent fuel discharge projections in Section 2.2.
2) The reprocessing facility starts operation in 2030 and begins at full capacity of:
   - Scenario 1 - 1,500 MT/year
   - Scenario 2 - 3,000 MT/year
3) Fuel must be at least 5 years old for reprocessing and fuel selection will start with youngest fuel first.
4) Only PWR fuel fabricated from new uranium is reprocessed, and none is disposed of in the repository. All other spent fuel is disposed of in the repository.
5) PWR MOX assemblies are fabricated from separated plutonium and fresh uranium tails (^{235}U assay in tails mass is 0.2%). MOX assemblies are limited to a maximum total plutonium content of 14%. No BWR MOX assemblies are fabricated.
6) PWR recycled UOX assemblies are fabricated from enriched recycled uranium (no blending of highly enriched uranium with the separated uranium). There is no limit on the maximum ^{235}U assay in the recycled UOX assemblies to off-set the loss of reactivity because of ^{236}U content. No BWR recycled UOX assemblies are fabricated.
7) There is an unlimited amount of natural uranium, natural uranium enrichment capacity, and new uranium UOX assembly fabrication capacity.
8) The repository starts operation in 2040 and begins at full capacity of 1,500 MT/year spent fuel. High level waste containing fission products and minor actinides is disposed of in the same repository, and in the same year that separation takes place, but with no limit on disposal capacity.
9) Spent fuel must be at least 10 years old for repository disposal.

2.5.2. Output Measures

Based on the assumptions in Section 2.5.1, calculate the following at the end of year 2100:

1) Total mass of PWR spent fuel disposed of in the repository.
2) Total mass of BWR spent fuel disposed of in the repository.
3) Mass of fission products and minor actinides, either total or by isotope, disposed of in the repository.
4) Total mass of PWR spent fuel reprocessed.
5) Percent reduction in total natural uranium demand.
6) Either total number or mass, and isotopic composition, of assemblies fabricated:
   - New uranium PWR assemblies
   - New uranium BWR assemblies
   - PWR recycled UOX assemblies (including ^{235}U assay)
   - PWR MOX assemblies (including ^{235}U assay, Pu quality, Pu percent)
7) Mass of uranium tails generated:
   - New uranium tails
   - Recycled uranium tails

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