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

Background

The Nuclear Waste Technical Review Board (NWTRB) held a Workshop on the Evaluation of Waste Streams Associated with LWR Fuel Cycle Options in Arlington, VA on June 6-7, 2011. The objectives of this workshop were to: 1) benchmark respective fuel cycle models/codes/tools, 2) establish consistency in input assumptions for the calculation of U.S. spent fuel generation and management, 3) understand how the scenario definitions provided by the NWTRB are applied in the calculation of spent fuel characteristics, and 4) reach consensus on areas of agreement, differences, and suggestions for future interactions and direction. The scenarios provided were not intended to be realistic representations of U.S. system operations, nor was the intent of the workshop to identify preferred scenarios.

Five scenarios were defined, sequenced as follows:

1. Determine the characteristics of the U.S. spent fuel inventory as of December 2009.
2. Calculate the quantity and composition of spent fuel discharged through 2100 for a specified electricity production and reactor burn-up.
3. Assess the impact of repository disposal with respect to the total mass of spent fuel disposed each year through 2100.
4. Estimate the steady-state impact on mass flows due to reprocessing and fabrication of pressurized water reactor (PWR) mixed oxide (MOX) and recycled uranium oxide (UOX) fuel.
5. Determine the impacts of reprocessing combined with repository disposal.

Performance measures of interest included the total mass of spent fuel generated, assemblies discharged, waste stream compositions, mass of new fuels generated, and the reduction in uranium demand realized.

The workshop was attended by "participants", defined as those organizations who utilized programs to perform analysis of the defined scenarios and present results at the workshop. In addition to the NWTRB, there were four other participants: AREVA, Idaho National Laboratory (INL), Massachusetts Institute of Technology (MIT), and the UK National Nuclear Laboratory (NNL)[1]. Representatives of other organizations and members of the public attended the workshop as "observers". These attendees were offered an opportunity to speak during structured discussion periods as well as during time set aside at the end of each day for public comment. A transcript of this meeting is posted on the NWTRB website (www.nwtrb.gov).

[1] Contact information for the four other participant organizations is as follows: AREVA – Paul Murray, paul.murray@areva.com; INL – Steven Piet, steven.piet@inl.gov; MIT – Stefano Passerini, stefanop@mit.edu; NNL – Robert Gregg; robert.wh.gregg@nnl.co.uk

Participant Analysis Tools

NUWASTE
The Nuclear Waste Assessment System for Technical Evaluation (NUWASTE) has been developed by the NWTRB as a material balance tool designed to assess the waste management implications of alternative fuel cycle scenarios for the existing and planned fleet of U.S. light-water reactor (LWR) nuclear plants. On the basis of assumed fuel burn-ups and initial uranium enrichments, NUWASTE calculates the masses of individual isotopes from input recipes in spent fuel assemblies that have been, or will be, discharged by the reactor fleet. The isotope masses in spent nuclear fuel are calculated by the ORIGEN code. Reprocessing scenarios are not based on any particular separations process, but the degree of separations can be specified. Based on the quantities of materials in various product streams, the amount of recycle fuel that can be fabricated is calculated for loading into reactors on a one-year time step basis. NUWASTE keeps track of all masses, assemblies and waste packages, including the composition of these streams. The tool is self-contained and pertinent data required for material balance calculations are obtained from externally-generated lookup tables.

AREVA
The analysis tool used by AREVA was a steady-state spreadsheet material flow calculation specific to the reprocessing and fabrication of PWR MOX and recycled UOX fuel. All data are based on operational experience at the La Hague recycling plant and the MELOX MOX fabrication plant. The isotopic calculations for fission products, actinides and activation products utilize CESAR data. The analysis takes into account reprocessing capacity, recycled fuel as PWR UOX with a specified initial enrichment from natural uranium derived fuel, discharged fuel cooling times, legacy fuel characteristics, and a fleet spent-fuel discharge rate corresponding to an electric generation capacity of 100.3 GWe.

VISION
INL has developed VISION, Verifiable Fuel Cycle Simulation, as a tool for evaluating advanced fuel cycle options. The code represents uranium milling and mining, conversion and enrichment, fuel fabrication as a function of energy demand, thermal and fast reactors, reprocessing/separations for spent fuel, and product recycle back to fuel fabrication. Low-level, greater than Class C, transuranic and high-level waste are considered. Material flow is dependent on energy demand. Up to 10 reactor types can be evaluated, representing either light-water or fast reactors, and material flow is routed from reactors to separations, from which fuel values can be recycled or disposed. VISION is not designed to analyze hundreds of reactors, has a single “legacy” retirement profile that begins in 1960 so that existing reactors can retire on time, obtains results from an as-stable-as-possible portion of a simulation rather than a true steady state, incorporates input/output fuel recipes rather than performing reactor physics, and considers only mass flow (not fuel assemblies).

CAFCA
MIT’s nuclear fuel cycle code is CAFCA, Code for Advanced Fuel Cycles Assessment, a systems analysis tool used to produce results reported in *The Future of Nuclear Fuel Cycle* study, published in April 2011. The code was constructed to assess alternative nuclear fuel cycles and corresponding impacts in the context of the U.S. energy scenario. CAFCA receives as inputs an

energy demand growth rate to be covered by nuclear power plants and the fuel cycle strategy, along with the related reactor technologies that are available for the considered scenario. Several metrics of interest are produced, including installed capacities over time, nuclear waste streams, and economics. Several fuel cycle and reactor technologies are resident within CAFCA. To analyze the workshop scenarios, the LWR once-through and twice-through cycles were selected. CAFCA is a discrete-time code, tracking mass flows of material. Only equilibrium core compositions are considered and no explicit distinction is made by the code between PWRs and BWRs. However, the cumulative mass flows can be separated (using EXCEL or any other data-processing software) once the simulation is completed if the relative composition of the LWR fleet is known. No isotope tracking is currently implemented in CAFCA and spent fuel vectors are calculated using fixed composition vectors. Combined use of CASMO (for burn-up calculations) and CAFCA allowed the expression of results according to the metrics specified in the workshop scenarios and to apply different spent fuel composition vectors to the mass flows estimated by CAFCA.

ORION

NNL's fuel cycle model, named ORION, can track up to 2,500 nuclides as nuclear material is moved through a fuel cycle. The smallest time step that can be defined is one year, thus parameters such as irradiation time for reactor fuel and reprocessing lead times must be an integer number of years. ORION considers six objects, or process steps: 1) reactor(s), 2) fuel-fabrication plant, 3) buffer, 4) active plant, 5) passive plant and 6) external feed. ORION will decay material as it flows around a fuel cycle scenario and will calculate the spent fuel inventory discharged from a reactor directly using cross sections and neutron fluxes generated by a reactor physics code such as CASMO-4, WIMS or ECCO. Performing the transmutation calculation in such a way allows ORION to model complex scenarios where the input fuel composition varies over time (i.e., closed cycles involving fast reactors and MOX utilization in LWRs). These objects can be "dragged" into the analysis and linked together as specified by the user. The current version of ORION does not consider a preference for the reprocessing of spent fuel with respect to time in/out of storage; material entering storage is mixed with the existing inventory. However, the program can preferentially process particular streams of material. In analyzing the workshop scenarios, present and future spent fuel were divided into separate streams by considering spent fuel discharged from the current reactor fleet before and after 2010, as well as new spent fuel from the new-build fleet. Due to the time step limitation of one year, the dwell time of fuel in a reactor must be an integer number of years, which in turn could limit burn-ups that can be considered. To circumvent this, varying the input parameter of core mass can be used to achieve a specified burn-up. For the purpose of this benchmark, individual reactors were not modeled; rather individual reactors were grouped to form reactor 'units' pertaining to PWRs and BWRs.

Analysis Results

Appendix A presents the defined scenarios for which analysis results were to be produced. Appendix B displays the workshop analysis results, organized by scenario, as produced by each of the participants. Note that some participants did not produce results for certain scenarios, due to restrictions in utilizing their tools to represent a certain scenario or to produce the required performance measures.

Although a variety of technical approaches were used by participants to analyze the scenarios provided, the majority of the results are in general agreement. This suggests that there is a reasonable degree of consistency in the underlying methodology of various organizations for evaluating fuel cycle options in terms of the waste streams and waste forms generated by different LWR scenarios.

One interesting observation is the sensitivity of isotopic compositions to burn-up codes. An example of this circumstance appears in the PWR results for Pu-241 in Scenario 2.2: Spent Fuel Discharged Through 2100. Here, the masses estimated by NWTRB, NNL, INL and MIT are 54.4, 30.8, 133.5 and 277.8 MT, respectively. It is important to recognize that Pu-241 is at the end of a neutron capture and thus different cross-sections for neutron capture by Pu-240 will affect this calculation. Note, however, that the Pu-240 results are quite consistent in this case, suggesting that the burn-up code for each respective estimation approach is more likely to be decaying Pu-241 differently. To clarify these inconsistencies, a benchmarking of burn-up codes would be a worthwhile endeavor.

Another area that proved challenging is estimation of the mass of specific isotopes discharged in BWRs. This can be attributed to difficulties in predicting average spent fuel discharge compositions for BWRs. Unlike PWRs the moderator density can vary significantly since water is allowed to boil and thus affects the neutron spectrum. Therefore, the cross sections do vary along the length of an assembly adding to the complexity of calculating an average composition of the discharged fuel. A benchmark of burn-codes as applied to a BWR to reconcile these differences might also be a worthwhile exercise

Finally, additional attention should be devoted to calculating the separated masses of UOX and MOX resulting from various combinations of reprocessing capacity and fuel age at the time of reprocessing.

Conclusions

Participants and observers agreed that the workshop served an important purpose by bringing together stakeholders to discuss the manner in which waste management implications of various fuel cycle options are evaluated. As a result, the workshop was viewed as a catalyst in the development of standard assumptions, parameters and methods for the generation and disposition of wastes so that all interested parties are “speaking the same language”. Future activities will hopefully be undertaken to further this pursuit.

Appendix A

Workshop Scenario Definitions

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. Scenarios

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 offset 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 offset 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

Appendix B

Workshop Results by Scenarios

Workshop Results for Scenario 2.1: Characteristics of U.S. Spent Fuel Inventory as of December 2009

Total Mass of Spent Fuel: Beginning of 2010							
Item	Number of Assemblies			Mass of Assemblies (MT)			Check
	PWR	BWR	Total	PWR	BWR	Total	
NWTRB	94,289	117,694	211,983	40,544.3	21,184.9	61,729.2	61,729.2
NNL	94,400	117,245	211,645	40,591.8	21,104.2	61,696.0	61,624.8
INL			212,021	38,948.0	22,118.0	61,066.0	
MIT	94,289	117,693	211,983	40,617.0	21,104.0	61,721.0	61,725.1
AREVA							

Mass of U-234, U-235, U-236 and U-238 in Spent Fuel: Beginning of 2010						
Item	PWR Masses (MT)					
	U-234	U-235	U-236	U-238	Total	Enrich
NWTRB	6.7	327.3	187.3	37,934.8	38,456.1	0.85%
NNL	7.5	293.5	178.1	38,041.0	38,520.0	0.76%
INL	0.5	267.3	157.8	36,622.8	37,048.4	0.72%
MIT	6.0	308.5	181.4	38,055.4	38,551.2	0.80%
AREVA						

Mass of U-234, U-235, U-236 and U-238 in Spent Fuel: Beginning of 2010						
Item	BWR Masses (MT)					
	U-234	U-235	U-236	U-238	Total	Enrich
NWTRB	2.6	73.6	70.0	20,147.7	20,293.8	0.36%
NNL	2.6	85.9	65.5	20,062.0	20,216.1	0.42%
INL	0.2	151.8	79.5	20,921.1	21,152.6	0.72%
MIT	2.3	95.9	68.4	20,063.1	20,229.7	0.47%
AREVA						

Total Mass of U-234, U-235, U-236 and U-238 in Spent Fuel: Beginning of 2010						
Item	Total (PWR + BWR) Masses (MT)					
	U-234	U-235	U-236	U-238	Total	Enrich
NWTRB	9.3	400.8	257.3	58,082.5	58,749.9	0.68%
NNL	10.1	379.4	242.7	58,103.0	58,735.2	0.65%
INL	0.7	419.1	237.3	57,543.9	58,201.0	0.72%
MIT	8.3	404.4	249.8	58,118.5	58,780.9	0.69%
AREVA						

Mass of Pu-238, Pu-239, Pu-240, Pu-241 & Pu-242 in Spent Fuel: Beginning of 2010							
Item	PWR Masses (MT)						
	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Total	Quality
NWTRB	7.4	230.2	104.7	29.7	27.2	399.3	65.1%
NNL	6.2	226.3	105.9	30.7	24.8	393.9	65.3%
INL	6.4	205.1	96.3	33.4	23.7	364.9	65.4%
MIT	7.5	224.9	98.5	60.5	25.4	416.8	68.5%
AREVA							

Mass of Pu-238, Pu-239, Pu-240, Pu-241 & Pu-242 in Spent Fuel: Beginning of 2010							
Item	BWR Masses (MT)						
	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Total	Quality
NWTRB	1.9	79.9	51.0	9.9	13.7	156.4	57.4%
NNL	2.0	98.4	50.1	12.3	12.0	174.9	63.3%
INL	2.8	112.9	51.5	17.0	11.4	195.6	66.4%
MIT	2.6	87.5	49.6	22.4	12.2	174.2	63.1%
AREVA							

Total Mass of Pu-238, Pu-239, Pu-240, Pu-241 & Pu-242 in Spent Fuel: Beginning of 2010							
Item	Total (PWR +BWR) Masses (MT)						
	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Total	Quality
NWTRB	9.3	310.1	155.7	39.6	40.9	555.7	62.9%
NNL	8.2	324.7	156.0	43.1	36.8	568.8	64.7%
INL	9.2	318.0	147.8	50.4	35.1	560.5	65.7%
MIT	10.0	312.4	148.1	82.9	37.6	591.0	66.9%
AREVA							

Mass of Fission Products & Minor Actinides, Total or by Isotope: Beginning of 2010			
Item	PWR Masses (MT)	BWR Masses (MT)	Total Masses (MT)
	FP & Minor Actinides	FP & Minor Actinides	FP & Minor Actinides
NWTRB	1,688.9	734.7	2,423.6
NNL	1,627.6	693.2	2,320.8
INL	1,534.4	769.3	2,303.7
MIT	1,651.9	701.3	2,353.2
AREVA			

Workshop Results for Scenario 2.2:
Spent Fuel Discharged Through 2100

PWR and BWR Assemblies in Storage Prior to 2010						
Item	Number of Assemblies			Mass of Assemblies (MT)		
	PWR	BWR	Total	PWR	BWR	Total
NWTRB	94,289	117,694	211,983	40,544	21,185	61,729
NNL	94,399	117,245	211,644	40,592	21,104	61,696
INL			212,021	38,948	22,118	61,066
MIT	94,289	117,694	211,983			
AREVA						

PWR and BWR Assemblies Generated from 2010 Through 2100						
Item	Number of Assemblies			Mass of Assemblies (MT)		
	PWR	BWR	Total	PWR	BWR	Total
NWTRB	274,876	353,824	628,700	118,197	63,688	181,885
NNL	266,968	336,164	603,132	114,796	60,509	175,306
INL			639,032	117,488	66,565	184,053
MIT	194,326	467,887	662,213			
AREVA						

Total PWR and BWR Assemblies Through 2100							
Item	Number of Assemblies			Mass of Assemblies			Check
	PWR	BWR	Total	PWR	BWR	Total	
NWTRB	369,165	471,518	840,683	158,741	84,873	243,614	243,614
NNL	361,367	453,409	814,776	155,388	81,614	237,001	236,408
INL			851,053	156,436	88,683	245,119	
MIT	288,615	585,581	874,196	124,104	105,405	229,509	239,335
AREVA							

Mass of U-234, U-235, U-236 and U-238 Discharged						
Item	PWR Masses (MT)					
	U-234	U-235	U-236	U-238	Total	Enrich
NWTRB	27.5	1,197.6	928.2	146,179.4	148,332.7	0.81%
NNL	44.4	1,345.4	242.7	143,495.2	145,127.7	0.93%
INL	5.7	1,114.0	870.0	144,359.6	146,349.3	0.76%
MIT	26.0	1,193.3	954.4	146,195.1	148,368.7	0.80%
AREVA						

Mass of U-234, U-235, U-236 and U-238 Discharged						
Item	BWR Masses (MT)					
	U-234	U-235	U-236	U-238	Total	Enrich
NWTRB	13.9	314.0	475.1	78,842.9	79,645.9	0.39%
NNL	20.4	497.1	413.7	76,010.4	76,941.5	0.65%
INL	3.1	631.5	483.1	81,960.6	83,078.3	0.76%
MIT	12.0	420.2	453.1	74,924.6	75,809.9	0.55%
AREVA						

Total Mass of U-234, U-235, U-236 and U-238 Discharged						
Item	Total (PWR + BWR) Masses (MT)					
	U-234	U-235	U-236	U-238	Total	Enrich
NWTRB	41.5	1,511.7	1,403.2	225,022.2	227,978.6	0.66%
NNL	64.8	1,842.5	656.3	219,505.6	222,069.2	0.83%
INL	8.8	1,745.5	1,353.1	226,320.2	229,427.6	0.76%
MIT	38.0	1,613.5	1,407.5	221,119.7	224,178.6	0.72%
AREVA						

Mass of Pu-238, Pu-239, Pu-240, Pu-241 and Pu-242 Discharged							
Item	PWR Masses (MT)						
	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Total	Quality
NWTRB	53.6	950.7	462.0	54.4	148.3	1,669.0	60.2%
NNL	28.8	968.0	443.0	30.8	126.3	1,596.8	62.5%
INL	45.3	896.9	448.5	133.5	136.0	1,660.2	62.1%
MIT	52.0	931.2	434.7	277.8	140.0	1,835.7	65.9%
AREVA							

Mass of Pu-238, Pu-239, Pu-240, Pu-241 and Pu-242 Discharged							
Item	BWR Masses (MT)						
	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Total	Quality
NWTRB	20.6	334.9	226.5	18.4	79.5	679.8	52.0%
NNL	10.4	381.9	216.6	10.4	60.7	680.0	57.7%
INL	25.0	504.9	251.0	75.4	75.0	931.3	62.3%
MIT	22.9	361.2	220.8	105.3	70.9	781.2	59.7%
AREVA							

Total Mass of Pu-238, Pu-239, Pu-240, Pu-241 and Pu-242 Discharged							
Item	Total (PWR + BWR) Masses (MT)						
	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Total	Quality
NWTRB	74.2	1,285.6	688.5	72.8	227.8	2,348.8	57.8%
NNL	39.2	1,349.8	659.6	41.2	187.0	2,276.8	61.1%
INL	70.3	1,401.8	699.5	208.9	211.0	2,591.5	62.2%
MIT	75.0	1,292.4	655.6	383.0	211.0	2,616.9	64.0%
AREVA							

Mass of Fission Products and Minor Actinides Discharged, Either Total or by Isotope			
Item	FP & Minor Actinides (MT)		
	In PWR Assemblies	In BWR Assemblies	Total
NWTRB	8,739	4,547	13,287
NNL	8,070	3,992	12,062
INL	8,426	4,673	13,100
MIT	8,425	4,115	12,540
AREVA			

Workshop Results for Scenario 2.3: Impact of Repository Disposal

Total Mass of PWR and BWR Disposed Through Year 2100 – Repository Capacity of 1,500 MT/year						
Item	Assemblies Disposed			MTU Disposed		
	PWR	BWR	Total	PWR	BWR	Total
NWTRB	138,285	175,164	313,449	59,463	31,530	90,992
NNL	131,922	207,040	338,962	56,727	37,267	93,994
INL			317,688			91,500
MIT				60,460	30,540	91,000
AREVA						

Total Mass of PWR and BWR Disposed Through Year 2100 – Repository Capacity of 3,000 MT/year						
Item	Assemblies Disposed			MTU Disposed		
	PWR	BWR	Total	PWR	BWR	Total
NWTRB	277,137	351,560	628,697	119,169	63,281	182,450
NNL	323,086	273,065	596,151	138,927	49,152	188,079
INL			688,963			198,434
MIT				110,845	55,989	166,834
AREVA						

Workshop Results for Scenario 2.4:
Steady State Reprocessing and Fabrication of
PWR MOX and Recycled UOX Fuel

Reprocessing Capacity of 1,500 MT/Year and All Fuel 5 Years Old						
Item	Mass FP & Minor Actinides (MT)	% Uranium Reduction	Natural Uranium			
			PWR UOX		BWR UOX	
			Number	Enrichment	Number	Enrichment
NWTRB	89	18.4%	2,151	4.40%	3,551	4.35%
NNL	90	20.8%	1,846	4.40%	3,288	4.34%
INL				4.59%		4.59%
MIT	87	15.2%	2,197	4.40%	3,636	4.40%
AREVA		18.0%	2,123	4.40%	3,483	4.40%

Reprocessing Capacity of 1,500 MT/Year and All Fuel 5 Years Old							
Item	Separated Mass					Mass Tails (MT)	
	PWR UOX		PWR MOX				
	Number	Enrichment	Number	% Pu	Pu Quality	Fresh	Separated
NWTRB	399	5.0%	416	10.03%	61.85%	10,996	1,221
NNL	422	5.1%	458	9.90%	64.10%	9,947	1,296
INL				10.60%	64.30%		
MIT	349	5.0%	467	8.73%	63.80%	11,596	1,388
AREVA	326	5.0%	312	14.00%	63.20%	12,548	1,252

Reprocessing Capacity of 1,500 MT/Year and All Fuel 5 Years Old			
Item	Total PWR Assemblies	Recycled PWR Assemblies	% Recycled Assemblies
NWTRB	2,966	815	27.5%
NNL	2,725	880	32.3%
INL			
MIT	3,013	816	27.1%
AREVA	2,760	637	23.1%

Reprocessing Capacity of 1,500 MT/Year and All Fuel 25 Years Old						
Item	Mass FP & Minor Actinides (MT)	% Uranium Reduction	Natural Uranium			
			PWR UOX		BWR UOX	
			Number	Enrichment	Number	Enrichment
NWTRB	89	16.6%	2,231	4.40%	3,551	4.35%
NNL	90	18.8%	1,931	4.40%	3,288	4.34%
INL				4.59%		4.59%
MIT	88	14.7%	2,234	4.40%	3,636	4.40%
AREVA		16.0%	2,228	4.40%	3,483	4.40%

Reprocessing Capacity of 1,500 MT/Year and All Fuel 25 Years Old							
Item	Separated Mass					Mass Tails (MT)	
	PWR UOX		PWR MOX				
	Number	Enrichment	Number	% Pu	Pu Quality	Fresh	Separated
NWTRB	399	5.0%	336	11.50%	59.50%	10,950	1,221
NNL	422	5.1%	373	11.36%	62.09%	10,211	1,296
INL				10.60%	64.30%		
MIT	349	5.0%	430	8.73%	60.80%	11,711	1,388
AREVA	326	5.0%	286	14.00%	60.20%	12,929	1,254

Reprocessing Capacity of 1,500 MT/Year and All Fuel 25 Years Old			
Item	Total PWR Assemblies	Recycled PWR Assemblies	% Recycled Assemblies
NWTRB	2,966	735	24.8%
NNL	2,725	795	29.2%
INL			
MIT	3,013	779	25.9%
AREVA	2,840	612	21.5%

Reprocessing Capacity of 1,500 MT/Year and All Fuel 50 Years Old						
Item	Mass FP & Minor Actinides (MT)	% Uranium Reduction	Natural Uranium			
			PWR UOX		BWR UOX	
			Number	Enrichment	Number	Enrichment
NWTRB	89	15.8%	2,266	4.40%	3,551	4.35%
NNL	90	18.1%	1,967	4.40%	3,288	4.34%
INL				4.59%		4.59%
MIT	89	14.4%	2,250	4.40%	3,636	4.40%
AREVA		15.0%	2,263	4.40%	3,483	4.40%

Reprocessing Capacity of 1,500 MT/Year and All Fuel 50 Years Old							
Item	Separated Mass					Mass Tails (MT)	
	PWR UOX		PWR MOX				
	Number	Enrichment	Number	% Pu	Pu Quality	Fresh	Separated
NWTRB	399	5.0%	301	12.36%	58.37%	11,345	1,221
NNL	423	5.2%	336	12.19%	60.80%	10,324	1,296
INL				10.60%	64.30%		
MIT	350	5.0%	413	8.73%	59.50%	11,763	1,388
AREVA	326	5.0%	274	14.00%	58.90%	13,057	1,254

Reprocessing Capacity of 1,500 MT/Year and All Fuel 50 Years Old			
Item	Total PWR Assemblies	Recycled PWR Assemblies	% Recycled Assemblies
NWTRB	2,966	700	23.6%
NNL	2,725	758	27.8%
INL			
MIT	3,013	763	25.3%
AREVA	2,863	600	21.0%

Reprocessing Capacity of 3,000 MT/Year and All Fuel 5 Years Old						
Item	Mass FP & Minor Actinides (MT)	% Uranium Reduction	Natural Uranium			
			PWR UOX		BWR UOX	
			Number	Enrichment	Number	Enrichment
NWTRB	178	36.8%	1,335	4.40%	3,551	4.35%
NNL	180	42.4%	966	4.41%	3,288	4.34%
INL				4.59%		4.59%
MIT	173	30.3%	1,381	4.40%	3,636	4.40%
AREVA		34.0%	1,449	4.40%	3,483	4.40%

Reprocessing Capacity of 3,000 MT/Year and All Fuel 5 Years Old							
Item	Separated Mass					Mass Tails (MT)	
	PWR UOX		PWR MOX				
	Number	Enrichment	Number	% Pu	Pu Quality	Fresh	Separated
NWTRB	798	5.0%	833	10.03%	61.85%	8,515	2,441
NNL	844	5.2%	916	9.90%	64.10%	7,215	2,592
INL				10.60%	64.30%		
MIT	698	5.0%	933	8.73%	63.80%	9,053	2,777
AREVA	1,556	5.0%	593	14.00%	61.80%	10,057	2,505

Reprocessing Capacity of 3,000 MT/Year and All Fuel 5 Years Old			
Item	Total PWR Assemblies	Recycled PWR Assemblies	% Recycled Assemblies
NWTRB	2,966	1,631	55.0%
NNL	2,725	1,760	64.6%
INL			
MIT	3,013	1,632	54.2%
AREVA	3,597	2,149	59.7%

Reprocessing Capacity of 3,000 MT/Year and All Fuel 25 Years Old						
Item	Mass FP & Minor Actinides (MT)	% Uranium Reduction	Natural Uranium			
			PWR UOX		BWR UOX	
			Number	Enrichment	Number	Enrichment
NWTRB	179	33.1%	1,496	4.40%	3,551	4.35%
NNL	180	38.2%	1,136	4.40%	3,288	4.34%
INL				4.59%		4.59%
MIT	176	29.3%	1,454	4.40%	3,636	4.40%
AREVA		31.0%	1,567	4.40%	3,483	4.40%

Reprocessing Capacity of 3,000 MT/Year and All Fuel 25 Years Old							
Item	Separated Mass					Mass Tails (MT)	
	PWR UOX		PWR MOX				
	Number	Enrichment	Number	% Pu	Pu Quality	Fresh	Separated
NWTRB	798	5.0%	672	11.50%	59.50%	9,005	2,441
NNL	844	5.2%	745	11.36%	61.81%	7,743	2,592
INL				10.60%	64.30%		
MIT	699	5.0%	860	8.73%	60.80%	9,280	2,777
AREVA	1,561	5.0%	563	14.00%	59.70%	10,492	2,507

Reprocessing Capacity of 3,000 MT/Year and All Fuel 25 Years Old			
Item	Total PWR Assemblies	Recycled PWR Assemblies	% Recycled Assemblies
NWTRB	2,966	1,470	49.6%
NNL	2,725	1,590	58.3%
INL			
MIT	3,013	1,559	51.7%
AREVA	3,691	2,124	57.5%

Reprocessing Capacity of 3,000 MT/Year and All Fuel 50 Years Old						
Item	Mass FP & Minor Actinides (MT)	% Uranium Reduction	Natural Uranium			
			PWR UOX		BWR UOX	
			Number	Enrichment	Number	Enrichment
NWTRB	179	31.6%	1,566	4.40%	3,551	4.35%
NNL	180	36.8%	1,209	4.41%	3,288	4.34%
INL				4.59%		4.59%
MIT	177	28.9%	1,488	4.40%	3,636	4.40%
AREVA		30.0%	1,609	4.40%	3,483	4.40%

Reprocessing Capacity of 3,000 MT/Year and All Fuel 50 Years Old							
Item	Separated Mass					Mass Tails (MT)	
	PWR UOX		PWR MOX				
	Number	Enrichment	Number	% Pu	Pu Quality	Fresh	Separated
NWTRB	798	5.0%	602	12.36%	58.37%	9,217	2,441
NNL	845	5.1%	672	12.22%	60.91%	7,969	2,592
INL				10.60%	64.30%		
MIT	699	5.0%	826	8.73%	59.5%	9,386	2,777
AREVA	1,561	5.0%	549	14.00%	58.90%	10,647	2,507

Reprocessing Capacity of 3,000 MT/Year and All Fuel 50 Years Old			
Item	Total PWR Assemblies	Recycled PWR Assemblies	% Recycled Assemblies
NWTRB	2,966	1,400	47.2%
NNL	2,725	1,517	55.7%
INL			
MIT	3,013	1,525	50.6%
AREVA	3,719	2,110	56.7%

Workshop Results for Scenario 2.5: Impacts of Reprocessing Combined With Repository Disposal

Reprocessing Capacity of 1,500 MT/Year					
Item	Assemblies Disposed				
	PWR		BWR		Total Mass (MT)
	Number	Mass (MT)	Number	Mass(MT)	
NWTRB	43,482	18,697	403,260	72,587	91,284
NNL	107,780	46,345	248,700	44,766	91,111
INL					
MIT		19,043		71,957	91,000
AREVA					

Reprocessing Capacity of 1,500 MT/Year				
Item	Mass FP & Minor Actinides Disposed (MT)	PWR Processed		% Reduction Natural Uranium
		Number	Mass (MT)	
NWTRB	5,182	247,648	106,489	13.1%
NNL	5,221	205,808	88,498	11.3%
INL				
MIT	5,938		105,000	11.2 %
AREVA				

Reprocessing Capacity of 1,500 MT/Year				
Item	Assemblies Fabricated (after 2009)			
	Natural Uranium			
	PWR UOX		BWR UOX	
	Number	Average Enrichment	Number	Average Enrichment
NWTRB	220,520	4.40%	355,437	4.35%
NNL	218,620	4.45%	349,355	4.34%
INL				
MIT	242,688	4.40%	345,639	4.40%
AREVA				

Reprocessing Capacity of 1,500 MT/Year							
Assemblies Fabricated (after 2009)						Tails Mass (MT)	
Item	Separated Mass						
	PWR UOX		PWR MOX				
	Number	Average Enrichment	Number	% Pu	Pu Quality	Fresh	Recycled
NWTRB	28,645	4.97%	26,828	10.35%	61.45%	1,106,111	85,469
NNL	33,482	5.10%	23,547	9.33%	64.15%	1,134,434	68,300
INL							
MIT	22,784	5.00%	29,805	8.73%	63.80%	941,009	84,783
AREVA							

Reprocessing Capacity of 3,000 MT/Year					
Item	Assemblies Disposed				
	PWR		BWR		Total Mass (MT)
	Number	Mass (MT)	Number	Mass(MT)	
NWTRB	56,595	24,336	371,756	66,916	91,252
NNL	57,076	24,543	370,121	66,622	91,164
INL					
MIT		25,356		65,664	91,000
AREVA					

Reprocessing Capacity of 3,000 MT/Year				
Item	Mass FP & Minor Actinides Disposed (MT)	PWR Processed		% Reduction Natural Uranium
		Number	Mass (MT)	
NWTRB	5,809	293,013	125,996	15.6%
NNL	5,912	256,373	110,240	14.8%
INL				
MIT	7,344		130,300	14.3%
AREVA				

Reprocessing Capacity of 3,000 MT/Year				
Item	Assemblies Fabricated (after 2009)			
	Natural Uranium			
	PWR UOX		BWR UOX	
	Number	Average Enrichment	Number	Average Enrichment
NWTRB	209,884	4.40%	355,437	4.35%
NNL	208,722	4.38%	349,355	4.34%
INL				
MIT	229,965	4.40%	345,639	4.40%
AREVA				

Reprocessing Capacity of 3,000 MT/Year							
Assemblies Fabricated (after 2009)						Tails Mass (MT)	
Item	Separated Mass						
	PWR UOX		PWR MOX				
	Number	Average Enrichment	Number	% Pu	Pu Quality	Fresh	Recycled
NWTRB	35,087	4.92%	31,022	10.26%	61.81%	1,072,082	101,541
NNL	38,266	5.10%	28,655	9.38%	64.11%	1,094,510	87,700
INL							
MIT	32,714	5.00%	35,070	8.73%	63.80%	866,471	104,500
AREVA							