Calculation of Reactivity-Device Incremental Cross Sections Using MCNP Version 5

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ABSTRACT

This paper describes the logic that was implemented in a user-defined tally option TALLYX in MCNP-5 to calculate the scattering and the transport cross sections of reactivity devices. Incremental cross sections of the liquid-zone controllers and of the cobalt adjusters of a CANDU® reactor were calculated with the user-defined TALLYX subroutine and the standard tally cards. The MCNP-based incremental cross sections of the reactivity devices were used in reactor-core simulations. The results of the analysis showed that the average simulation error for the liquid-zone controllers was –7.4% for the MCNP-based incremental cross sections and –6.2% for the DRAGON-based Side-Step method. The methodology for using TALLYX in MCNP-5 is now considered ready for general use.

1. Introduction

The 3-dimensional transport code DRAGON [1, 2] is the recommended Industry Standard Toolset (IST) to calculate the incremental cross sections of reactivity devices. The incremental cross sections are used in the Reactor Fueling Simulation Program (RFSP) [3], a finite reactor code that uses multi-group diffusion theory to solve the neutronics of CANDU reactors. RFSP can calculate both static and kinetic neutron-flux and power distributions in the core in two energy groups and three dimensions. The presence of reactivity devices in the RFSP code is represented as an incremental change in cross sections in the mesh.

The DRAGON-IST code Version 3-03Bb is restricted to Cartesian and cylindrical geometries in modelling the reactivity devices in three dimensions. Hence the supercell calculations using the DRAGON code can only provide an approximation to the 3-dimensional cluster-type devices, which may perturb the neutron fine-flux distribution. The modelling approximation could have an effect on the incremental cross sections calculated for the reactivity devices.

This paper describes the use of the Monte Carlo transport code MCNP-5 [4, 5] as an alternate approach to calculate the incremental cross sections of reactivity devices. The MCNP-5 code was chosen to calculate the incremental cross sections because it has a more rigorous modelling capability and it uses a continuous cross-section data set; the incremental cross sections calculated would therefore be subject to less uncertainties and tend to be more accurate.

2. Supercell calculations

Each reactivity device modelled in the RFSP computer code is represented by eight incremental cross sections in the full-two-energy-group convention. Up to now, the incremental cross sections have been calculated using the DRAGON-IST supercell code by subtracting the cross sections calculated for a perturbed case from the cross sections calculated for a reference case. The reference case is a supercell that contains two fuel channels. The perturbed case is the same supercell with a reactivity device added perpendicular to the two fuel channels. The dimensions of the supercell model are two lattice pitches in width (57.15 cm) by one lattice pitch in height (28.575 cm), and one bundle length in the axial direction (49.53 cm).

The macroscopic incremental cross sections that are required for the reactivity devices in the RFSP calculation model are listed below.

$\Delta\Sigma_{a1}, \Delta\Sigma_{a2}$ = incremental absorption cross sections for the epithermal and thermal neutron-energy groups (cm^{-1})
$\Delta\nu\Sigma_{f1}, \Delta\nu\Sigma_{f2}$ = incremental fission-neutron-production cross sections for the epithermal and thermal neutron-energy groups (cm^{-1})
$\Delta\Sigma_{tr1}, \Delta\Sigma_{tr2}$ = incremental transport cross sections for the epithermal and thermal neutron-energy groups (cm^{-1})
$\Delta\Sigma_{1\rightarrow 2}$ = incremental down-scatter cross section from the epithermal neutron-energy group to the thermal neutron-energy group (cm^{-1})
$\Delta\Sigma_{2\rightarrow 1}$ = incremental up-scatter cross section from the thermal neutron-energy group to the epithermal neutron-energy group (cm^{-1})

Using MCNP-5, the same supercell model was set up, and the absorption and neutron-production cross sections were obtained using the standard F4 volume-averaged flux tally card. Because the in-group and out-of-group scattering cross-section and the transport cross-section data are not available in a standard MCNP-5 output, the user-defined TALLYX subroutine available in MCNP-5 was modified to extract the data to calculate the in-group and out-of-group scattering cross sections and the transport cross sections.

All MCNP-5 calculations were based on the continuous ENDF/B-VI data library. The cross sections of the reactivity devices were derived in a full-two-energy-group structure with an energy cutoff at 0.625 eV. The thermal group included neutrons located in the energy bin below the energy cutoff, and the epithermal neutron group included neutrons located in the energy bin above the energy cutoff up to 10 MeV. All results attained from the MCNP-5 supercell calculations were done with 2000 neutrons per cycle, 200 inactive cycles and a total of 2000 cycles.

3. Scattering and transport cross sections

A methodology has been developed to exploit the use of TALLYX in MCNP-5 to obtain the different reaction rates for the in-group and out-of-group neutron scattering collisions and the reaction rates resulting from the neutron transport collisions. The principle of the methodology is to monitor the collisions of each neutron during the transport process and verify that the neutron is within the volume of interest specified by the user. For neutrons that are within the volume of interest, all scatter events within the volume will be recorded, or tallied.

3.1 Scattering cross sections

MCNP-5 provides a user-defined tally option TALLYX that can be used to access and process the data available within the source code. A standard module is also available to calculate the standard deviation associated with the values obtained by TALLYX. Two energy bins were used to tally the in-group and the out-of-group scattering reaction rates. The cross sections were calculated with Equation 1 where the reaction rates in the numerator of the equation were tallied using the TALLYX subroutine. The denominator of the equation was the product of the flux times the volume of each material within the region of interest. User bin number 1 and user bin number 2 of Table 1 list the data that was tallied in the user-defined TALLYX subroutine to calculate the scattering cross sections.

$$\Sigma_{x,(1,2)} = \frac{\sum_{i=1}^{j} V_i \times R_{x,(1,2)}}{\sum_{i=1}^{j} \phi_{i,(1,2)} \times V_i} \qquad (1)$$

Where

$\Sigma_{x,(1,2)}$ macroscopic cross section of type x weighted by the epithermal or the thermal neutron flux (cm^{-1})

$R_{x,(1,2)}$ reaction rate of type x being homogenized for the epithermal or the thermal neutron-energy groups (reactions.s^{-1})

$\phi_{i,(1,2)}$ epithermal or the thermal neutron flux in region i (n.cm^{-2}.s^{-1})

V_i volume of region i (cm^{3})

3.2 Transport cross sections

Equation 2 was used to calculate the transport cross sections for the thermal and for the epithermal neutron-energy groups. The total macroscopic cross sections were tallied with the F4 tally cards for each material located within the volume of interest. Equation 1 was used to calculate the total cross sections for the volume of interest.

$$\sum\nolimits_{tr,g}(\vec{r}) = \sum\nolimits_{total,g}(\vec{r}) - \langle \mu_g \rangle \sum\nolimits_{el,g}(\vec{r}) \qquad (2)$$

$\sum_{total,g}(\vec{r})$ total macroscopic cross section for the neutron-energy group g (cm^{-1})

$\langle \mu_g \rangle$ average cosine of the elastic collision for the neutron-energy group g

$\sum_{el,g}(\vec{r})$ elastic macroscopic cross section for the neutron-energy group g (cm^{-1})

The average cosine of an elastic collision for the thermal, and for the epithermal neutron-energy groups is required per elastic collision. MCNP-5 tallies the reaction rates per particle but must be re-normalized per elastic collision. Hence three sets of data are tallied. The first set is the average cosine of a collision resulting from an elastic collision. The value is averaged over the number of elastic collisions. The second set of data is the average number of elastic collisions per neutron. The third set of data is the elastic scattering reaction rates resulting from the elastic collisions. User bin number 3 to user bin number 8 of Table 1 list the data that are tallied in the user-defined TALLYX subroutine to calculate the transport cross sections. Equations 3 and 4 are used to calculate the average cosine per elastic collision for the epithermal and for the thermal neutron-energy groups respectively.

$$\langle\mu_1\rangle = \frac{\langle\mu_{1\to2}\rangle\phi_1 + \langle\mu_{1\to1}\rangle\phi_1}{\langle C_{1\to2}\rangle\phi_1 + \langle C_{1\to1}\rangle\phi_1} \tag{3}$$

$$\langle\mu_2\rangle = \frac{\langle\mu_{2\to2}\rangle\phi_2 + \langle\mu_{2\to1}\rangle\phi_2}{\langle C_{2\to2}\rangle\phi_2 + \langle C_{2\to1}\rangle\phi_2} \tag{4}$$

Where
$\langle\mu_{x\to y}\rangle$ = average elastic scattering cosine angle per particle going from the neutron-energy group x to the neutron-energy group y
$\langle C_{x\to y}\rangle$ = average number of elastic collisions per particle going from the neutron-energy group x to the neutron-energy group y
$\langle\mu_x\rangle$ = average cosine of the elastic collisions for the neutron-energy group x

The values of μ and C are all tallied by the user-defined TALLYX subroutine within the region of interest specified by the user in the input file.

4. Verification of the user-defined tallyx subroutine

The user-defined TALLYX subroutine went through a verification process where the source code was independently reviewed to make sure that the methodology has been implemented properly.

The subroutine also tallies the elastic scattering reaction rates in user bins numbered 7 and 8. These bins can be compared directly against the reaction rates calculated using the standard F4 tally card. Generally, the difference between the user-defined tally bins of the elastic scattering reaction rates and the elastic scattering reaction rates calculated using the standard F4 tally card is within 0.5%. Hence it provides a self-check tool for the user to make sure that the volumes specified in the MCNP-5 input file have been properly defined.

5. Post simulations for the validation of the incremental cross sections

Two sets of measurements were selected to verify the accuracy of the incremental cross sections calculated with MCNP-5. The reactivity devices involved in the measurements were the Liquid-Zone Controllers (LZCs) and the cobalt adjusters. The next sections contain an overview of the information related to each set of measurements. The calculation model for the incremental cross sections are calculated based on these two measurement conditions.

5.1 Bruce A unit 4 phase B commissioning tests

The Bruce A Unit 4 reactor contains 480 fuel channels and each fuel channel contains 13 fuel bundles. Each bundle is a 37-element CANDU fuel bundle. The reactor has 14 LZCs that consist of light-water compartments. A water compartment is a cylinder that can be partially filled with light water. Changing the water level and therefore the neutron absorption rate in each light-water compartment controls the reactor power.

The procedure that was used to measure the reactivity worth of the LZCs was to start with a critical reactor, add batches of gadolinium, and lower the average fill levels of the LZCs to keep the reactor critical. During the measurements the poison addition was done in ppm gadolinium. In the commissioning report the values were transferred into mk using a reactivity coefficient of 30 mk/ppm. Table 3 gives the poison additions that were used during the calibration measurements of the LZCs. The validation of the incremental cross sections is to compare the relative reactivity change to the amount of gadolinium addition.

5.2 ROP perturbation tests for Gentilly-2

The Gentilly-2 (G2) reactor contains 380 fuel channels and each fuel channel contains 12 fuel bundles. Each bundle is a 37-element CANDU fuel bundle. The reactivity devices consist of 4 Mechanical Control Absorber rods (MCA), 28 Shutoff Rods (SOR), 14 LZCs, and 21 adjuster rods. The adjuster rods are usually inserted inside the core and one of their functions is to override the negative xenon reactivity following reactor shutdown or power increases.

In 2001, the Gentilly-2 (G2) reactor was shutdown for maintenance from April to early May and as part of the annual outage the stainless-steel adjusters were replaced by new cobalt adjusters. During the reactor startup, flux-map measurements were done and the values were used in HQSIMEX [6] to calculate the channel-power distribution, or the so-called pseudo-measured channel-power distribution. HQSIMEX is a computer code developed by Hydro-Québec. The fundamental mode that is used by the diffusion equation in HQSIMEX comes from the measured flux map, and therefore the calculated channel-power distribution is called a pseudo-measured channel-power distribution. Hence the error made on the calculated power distribution is of the second order. The incremental cross sections that were calculated with MCNP-5 were used in RFSP to calculate the channel-power distribution. The validation of the incremental cross sections using this approach is to compare the calculated channel-power distribution to the pseudo-measured channel-power distribution to determine the accuracy of the MCNP-based incremental cross sections used in the calculations.

6. Incremental cross sections

The next sections detail the methodology that was used to calculate the incremental cross sections of the reactivity devices.

6.1 Liquid-zone controllers

Two sets of tubes are located inside a LZC. The first set consists of a He balance line surrounded by a H_2O feed line. The second set consists of a He bubbler line surrounded by a H_2O scavenger line. Three supercell configurations were used to calculate the incremental cross sections of the LZCs. The configurations are a function of the number of tubes included in each compartment and they are listed below.

- Type 32: consists of 3 scavenger and 3 bubbler tubes, with 2 feeders and 2 balance tubes,
- Type 21: consists of 2 scavenger and 2 bubbler tubes, with 1 feeder and 1 balance tube,
- Type 10: consists of 1 scavenger and 1 bubbler tube, and no feeder or balance tube.

The incremental cross sections have been calculated for each type. The perturbed supercell configurations consisted of a LZC filled with light water and the reference case consisted of a LZC filled with ^4He. Hence the incremental cross sections represented the absorption effect of a LZC going from 0% to 100% filled. During normal operation the fill percentage is allowed to change in the range from 20% filled to 80% filled. For simplicity the effect of the absorption is traditionally calculated in the full-core model by multiplying the incremental cross sections by the fill fraction.

6.2 Cobalt adjusters

In G2, a cobalt bundle can contain one to four ^{59}Co pencils arranged on a 5.08 cm pitch circle. Note that when one cobalt pencil is used, a solid zirconium dummy pencil is placed symmetrically opposite for structural support. A cobalt pencil is comprised of eight cobalt slugs, each of 0.6223 cm in diameter and 2.5273 cm in length.

The standard CANDU supercell geometries were used to calculate the incremental cross sections of the cobalt bundles. The geometry consisted of two fuel channels with the reactivity device located in a perpendicular position between the two fuel channels. The dimensions of the supercell are two lattice pitches in width, one lattice pitch in height and one bundle length in the axial direction. Reflective boundary conditions were used along the X, Y, and the Z axes. The perturbed supercell configuration consisted of a supercell with a cobalt bundle between two fuel channels. The reference supercell configuration consisted of a guide tube located between two fuel channels. The guide tube was filled with heavy-water moderator.

The incremental cross sections were calculated at mid-burnup condition using WIMS-IST version 2-5d [7]. The ENDF/B-VI library was used for the analysis. The isotopic number densities that were calculated with WIMS-IST were used in MCNP-5 to model the fuel at mid-burnup condition.

7. Reactor-core simulations

The post simulations of the Bruce A Unit 4 Phase B commissioning tests and the ROP perturbation tests for Gentilly-2 using the MCNP-5 incremental cross sections were done differently. Hence the next two sections explain the methodology that was followed for each case.

7.1 Post simulations of the Bruce A unit 4 phase B commissioning tests

Table 4 lists the results of the reactor-core post simulations for the calibration of the liquid-zone controllers of the Bruce A Unit 4 Phase B commissioning tests. Table 4 shows that the average simulation error was –7.4%. Previous results show that the average simulation error of the core simulations based on the DRAGON Side-Step method for the incremental cross sections was –6.2%.

The results showed that the simulation error went from –5.71% at a 68.8% zone fill to -8.64% at a 13.2% zone fill. The trend suggests that the reactivity worth of the LZC was slightly over-estimated. This overestimation could be the result of modelling a slightly lower weight percent of gadolinium nitrate in the moderator therefore reducing the absorption rate in the moderator; this would result in a slight increase of the excess reactivity at higher concentrations of gadolinium nitrate. Another possible reason for the overestimation is on the use of supercell geometries to calculate the incremental cross sections instead of modelling a larger reactor core in MCNP-5. Hence the fine-flux distribution may be over-estimated in the supercell configuration resulting in a slight increase of the absorption rate.

The overestimation of the reactivity worth of the LZC could also be attributed to the use of the same set of incremental cross sections for all zone compartments in the reactor-core model; the incremental cross sections correspond to a zone fill level of 100%. Hence the behaviour of the incremental cross sections of the LZC in RFSP is assumed to be linear for a zone fill in the 0%-100% range. It may be that the deviation from linearity is related to the limitation to represent situations where the streaming through a highly voided region cannot be adequately modeled using a diffusion code.

7.2 Post simulations of the ROP perturbation tests for Gentilly-2

Full-core simulations of the Gentilly-2 reactor were done with the full 2-energy-group WIMS/DRAGON/MCNP/SCM/RFSP methodology up to 5148 Full Power Days (FPDs). At 5148 FPDs the full 2-energy-group MCNP-based incremental cross sections for the cobalt adjusters were substituted into the reactor-core model. The reactor-core model used the DRAGON-based 2-energy-group incremental cross sections for the liquid-zone controllers, the shutoff rods, the mechanical control absorbers, the guide tubes, the adjuster supporting bars and cables, the guide-tube brackets and locators, the tensioning springs and the coupling nuts.

The nominal power distribution at 5148 FPDs (just before reactor shutdown) was simulated with RFSP and the results were compared to the HQSIMEX simulations where the power distribution was calculated by mapping the 102 in-core vanadium detectors. The channel-power distribution and the bundle-power distribution at bundle plane number 6 that were calculated with RFSP were compared to the power distributions calculated with HQSIMEX. The results were shown in terms of relative difference between HQSIMEX and RFSP using Equation 5.

$$\text{Relative Percent Difference} = \left(P_{HQSIMEX} / P_{RFSP-IST} - 1\right) \times 100\% \quad (5)$$

Where

$P_{HQSIMEX}$ channel power distribution calculated using HQSIMEX (kW)

$P_{RFSP-IST}$ channel power distribution calculated using RFSP (kW)

Figure 1 shows the channel power comparison between HQSIMEX and the RFSP core simulations using the MCNP-based incremental cross sections for the cobalt adjusters. The channel power in HQSIMEX was calculated by mapping the 102 in-core vanadium detectors. Figure 1 shows that the use of the MCNP-based incremental cross sections for the cobalt adjusters in RFSP tends to slightly over-estimate the power distribution in the center of the core. Figure 1 also shows that the largest errors that were calculated with RFSP using the MCNP-based incremental cross sections for the cobalt adjusters were at the right-hand side and at the bottom of the reactor.

8. Conclusions

The use of the TALLYX subroutine in MCNP-5 has a significant advantage in calculating the incremental cross sections for the structural materials. With the TALLYX subroutine a full-core model can be used in MCNP-5 and the cross sections can be homogenized over a volume specified by the user and located anywhere in the model. The neutron spectrum calculated with MCNP-5 is a better representation of the neutron flux in the reactor. Hence the use of the TALLYX subroutine could also be used to calculate the incremental cross sections of the structural materials and the cross sections of the heavy-water moderator.

9. Acknowledgement

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10. References

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Table 1: Tallied Data in the User-Defined TALLYX Subroutine (Source Particle)

	Energy Bins	
User Bin Number	***Thermal***	***Epithermal***
1 Elastic & Non-Elastic Scatter	$\sum_{2\to 2}^{scat}\phi_2$	$\sum_{2\to 1}^{scat}\phi_2$
2 Elastic & Non-Elastic Scatter	$\sum_{1\to 2}^{scat}\phi_1$	$\sum_{1\to 1}^{scat}\phi_1$
3 Number of Elastic Collisions	$\langle C_{2\to 2}\rangle\phi_2$	$\langle C_{2\to 1}\rangle\phi_2$
4 Number of Elastic Collisions	$\langle C_{1\to 2}\rangle\phi_1$	$\langle C_{1\to 1}\rangle\phi_1$
5 Average Cosine of the Elastic Scatter	$\langle \mu_{2\to 2}\rangle\phi_2$	$\langle \mu_{2\to 1}\rangle\phi_2$
6 Average Cosine of the Elastic Scatter	$\langle \mu_{1\to 2}\rangle\phi_1$	$\langle \mu_{1\to 1}\rangle\phi_1$
7 Elastic Scatter	$\sum_{2\to 2}^{el.scat}\phi_2$	$\sum_{2\to 1}^{el.scat}\phi_2$
8 Elastic Scatter	$\sum_{1\to 2}^{el.scat}\phi_1$	$\sum_{1\to 1}^{el.scat}\phi_1$

Table 2: Reactor-Core Conditions for the Bruce A Unit 4 Phase B Commissioning Tests

Parameter	Unit	Value
Moderator Temperature	Celsius	25.0
Moderator Isotopic Purity	wt% D_2O	99.95
Moderator Density	g/cm^3	1.104
Coolant Temperature	Celsius	25.0
Coolant Isotopic Purity	wt% D_2O	99.1
Coolant Density	g/cm^3	1.101
Fuel Temperature	Celsius	25.0
Fuel Density	g/cm^3	10.61
Moderator Poison (Gd)	ppm	1.85

Table 3: Measured Average LZC Fills Versus Poison Addition (Bruce A Unit 4)

Poison addition (ppm Gadolinium)	Cumulative Poison Addition (ppm Gadolinium)	Measured Zone Level (% full)
0.0000	1.8500	77.7
0.0118	1.8618	68.8
0.0118	1.8736	60.5
0.0118	1.8854	53.0
0.0118	1.8972	45.8
0.0118	1.9090	38.7
0.0118	1.9208	31.6
0.0118	1.9326	24.3
0.0059	1.9385	20.4
0.0118	1.9503	13.2

Table 4: Post Simulations of the LZCs with the MCNP-Based Incremental Cross Sections

Measured Zone Level (%)	Poison Addition (mk)	Cumulative Poison Addition (mk)	Cumulative Predicted Reactivity Change (mk)	Relative Error % (Measured-Predicted)/ Measured
77.7	0.000	0.000	0.00	0.00
68.8	0.354	0.354	0.37	-5.71
60.5	0.354	0.708	0.76	-7.04
53	0.354	1.062	1.14	-7.55
45.8	0.354	1.416	1.52	-7.04
38.7	0.354	1.770	1.90	-7.34
316	0.354	2.124	2.28	-7.55
24.3	0.354	2.478	2.67	-7.66
20.4	0.177	2.655	2.87	-8.30
13.2	0.354	3.009	3.27	-8.64
			Average	-7.43

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	
A									0.77	-1.49	-1.62	-0.92	-1.18	1.04									A
B						-1.03	-2.61	-2.40	-2.76	-2.49	-2.27	-1.77	-1.77	-1.67	-0.95	-0.70	1.33						B
C					-2.96	-4.67	-3.38	-2.94	-2.36	-1.63	-1.47	-0.86	-1.12	-1.83	-1.62	-1.76	-1.38	0.80					C
D				-1.71	-4.37	-3.45	-2.82	-2.55	-1.80	-2.06	-0.73	0.29	0.03	-0.51	-0.76	-1.08	-0.34	-0.08	1.53				D
E			-1.05	-2.76	-4.67	-3.15	-2.05	-2.27	-1.53	-1.59	0.27	0.18	-0.47	0.39	-0.09	1.48	1.66	1.40	1.58	3.86			E
F			-2.19	-2.04	-2.13	-2.14	-2.30	-2.09	-1.89	-2.47	-0.80	-1.46	-1.89	-0.74	-0.16	1.24	1.44	1.73	0.77	2.79			F
G		-0.89	-2.91	-0.87	-1.65	-0.74	-0.88	-1.37	-3.02	-2.51	-1.49	-2.90	-1.47	-0.75	-0.78	1.91	1.15	2.39	2.11	2.43	4.30		G
H		-1.45	-2.45	-0.15	-0.72	-0.35	-1.18	-2.86	-2.34	-2.89	-1.94	-2.35	-3.38	-1.47	-0.64	1.65	2.08	3.70	2.01	3.27	3.07		H
J	0.91	-1.89	-1.65	0.99	-0.12	-1.59	-1.49	-2.60	-2.09	-1.69	-3.12	-1.31	-1.97	-0.43	-1.06	2.16	2.36	2.48	4.12	2.63	2.41	5.71	J
K	-0.33	-1.88	-1.26	0.84	-0.21	-0.65	-0.04	-1.18	-3.46	-2.67	-2.83	-1.78	-1.74	-1.81	1.05	1.77	3.92	4.04	3.66	2.74	1.95	3.51	K
L	0.07	-1.79	-0.58	0.52	0.47	-0.03	-0.42	-1.83	-1.82	-2.26	-3.20	-1.22	-1.54	-2.16	0.61	1.88	1.92	2.52	4.23	3.53	1.49	3.32	L
M	0.26	-1.34	0.06	1.15	0.12	-1.00	-1.09	-2.74	-2.75	-2.30	-3.54	-1.98	-1.01	-1.71	0.42	2.50	2.32	3.13	4.41	2.13	1.91	3.50	M
N	0.55	-1.04	-0.60	0.38	-0.23	-1.27	0.00	-0.32	-2.30	-1.36	-0.65	-1.90	-0.97	-1.27	1.12	0.63	1.42	2.80	3.84	3.53	1.76	3.50	N
O	0.93	-1.02	-0.89	1.58	1.93	1.12	-0.08	0.32	-1.15	-1.55	-1.49	-2.60	-1.61	0.68	0.61	2.49	2.89	3.76	2.47	2.70	1.29	4.56	O
P		-0.19	-0.55	1.30	1.77	1.97	1.27	-1.18	-0.89	0.03	0.35	0.37	-0.27	0.44	1.64	0.03	2.41	2.06	1.14	2.31	2.06		P
Q		0.44	-0.69	1.06	0.58	1.09	0.15	0.34	-0.80	-0.79	1.52	0.35	1.57	0.09	1.27	1.08	1.74	2.10	0.59	1.02	2.35		Q
R			0.08	0.56	-0.27	0.01	0.63	0.95	1.42	0.80	2.33	1.99	-0.85	1.77	1.35	2.27	2.14	1.21	-0.14	1.32			R
S			0.71	-0.13	-0.19	-0.11	0.30	0.80	0.84	0.53	1.86	2.49	1.80	1.49	1.49	0.66	0.36	0.50	-0.56	2.08			S
T				0.95	-1.33	-1.09	-0.41	0.11	0.08	0.36	0.55	1.86	1.70	1.44	0.26	0.42	-0.04	-0.43	-0.10				T
U					0.06	-1.16	-0.85	-0.70	-0.49	0.75	1.32	1.02	0.77	-0.06	0.22	-0.26	-0.52	0.76					U
V						1.61	0.32	0.14	-0.47	0.02	-0.02	0.51	0.50	0.37	0.93	1.09	1.97						V
W									4.41	1.61	1.85	1.93	2.41	4.76									W
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18	19	20	21	22	

Figure 1: Channel Power Comparison (%) HQSIMEX Versus RFSP with MCNP-Based Incremental Cross Sections (Cobalt Adjusters)