AXINT SAS :

AXINT2MCNP for MCNP file creation

ABOUT AXINT :

- AXINT is a spinoff from Lyon 1 University, LPCML laboratory.

-Axint was created in 2007.

-Main activity : Radiation detectors, mainly based on SiPM photodetectors.

-We propose also MCNP6/CAD radiation transport simulation studies.

-AXINT develops standart products for markets, and custom products for specific applications.

Required :

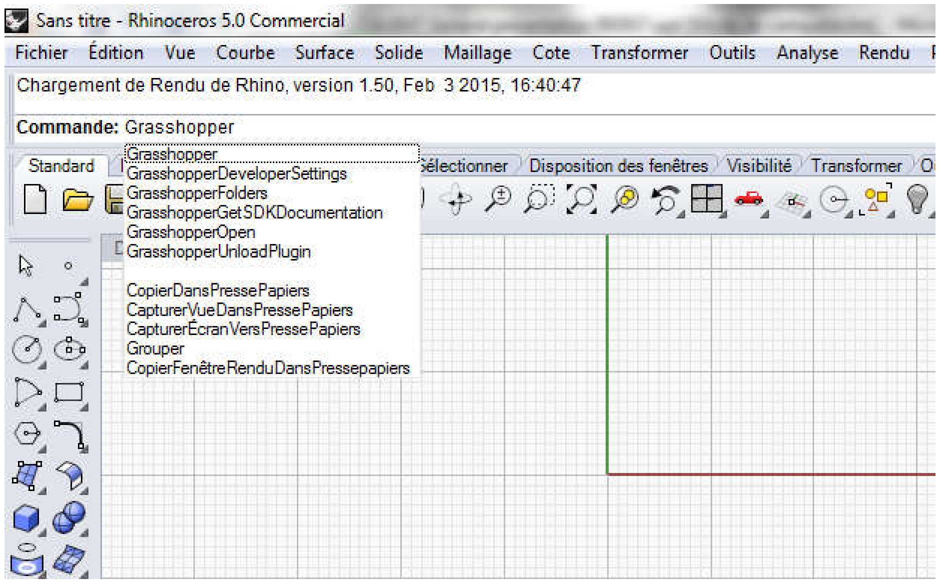
-Install Rhinoceros5 on the computer (evaluation version is enough). The software has NOT been tested with rhino 6 beta.

-Install grasshopper : http://www.grasshopper3d.com/

-HaveMCNP6.1 or earlier installed.

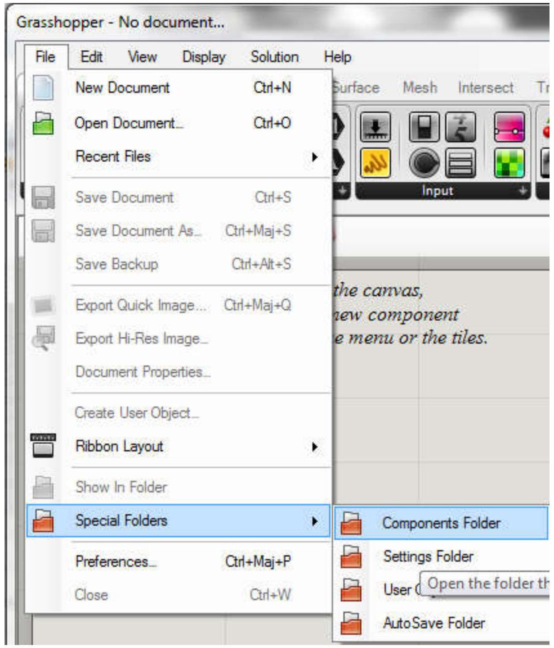
STEP 1: INSTALLATION OF AXINT_INP_WRITER

-Open Rhino, type the « grasshopper » command



STEP 1: INSTALLATION OF AXINT_INP_WRITER

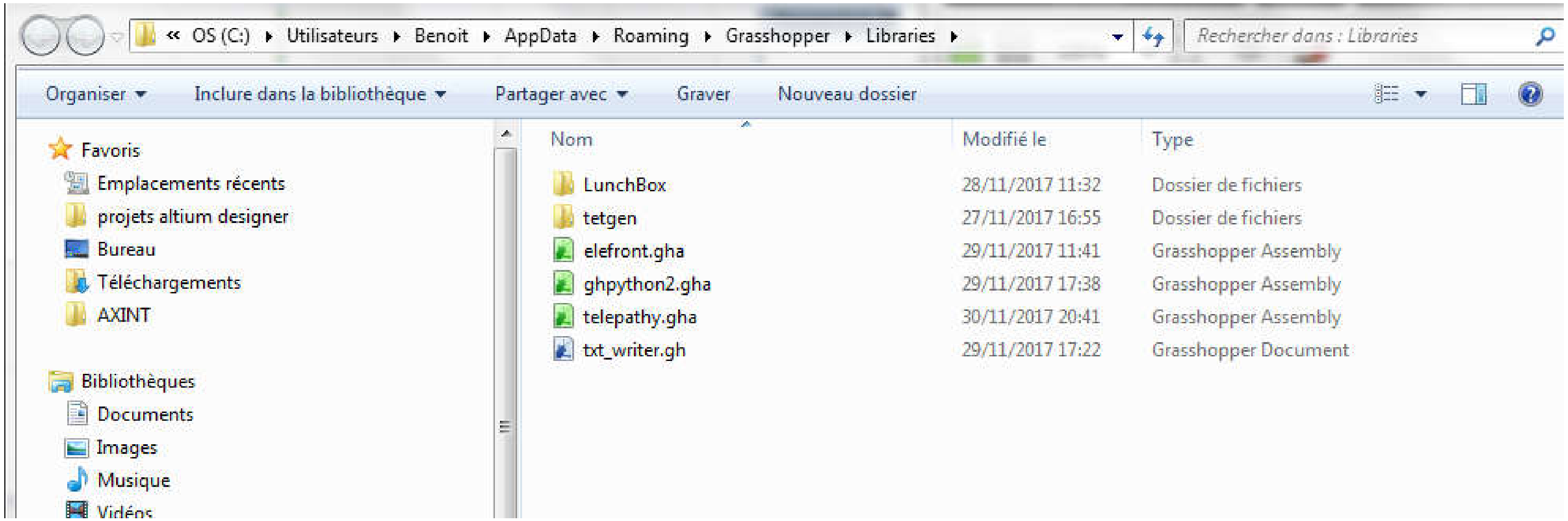
Grashopper will load, then : Files → special folders → components folder



STEP 1: INSTALLATION OF AXINT_INP_WRITER

-Past here the files from « grasshopper files » folder, Close Grasshopper and Rhino,

installation is ready.



STEP 1: INSTALLATION OF AXINT_INP_WRITER

-Create a folder where you want to do the files conversion, then copy here the

AXINT_INP_WRITER file here. Installation is ready.

STEP 2 : Preparing the CAD GEOMETRY

Prefer this option if you have no CAD already in your organisation

-Option 1 : Create your geometry with Rhino.

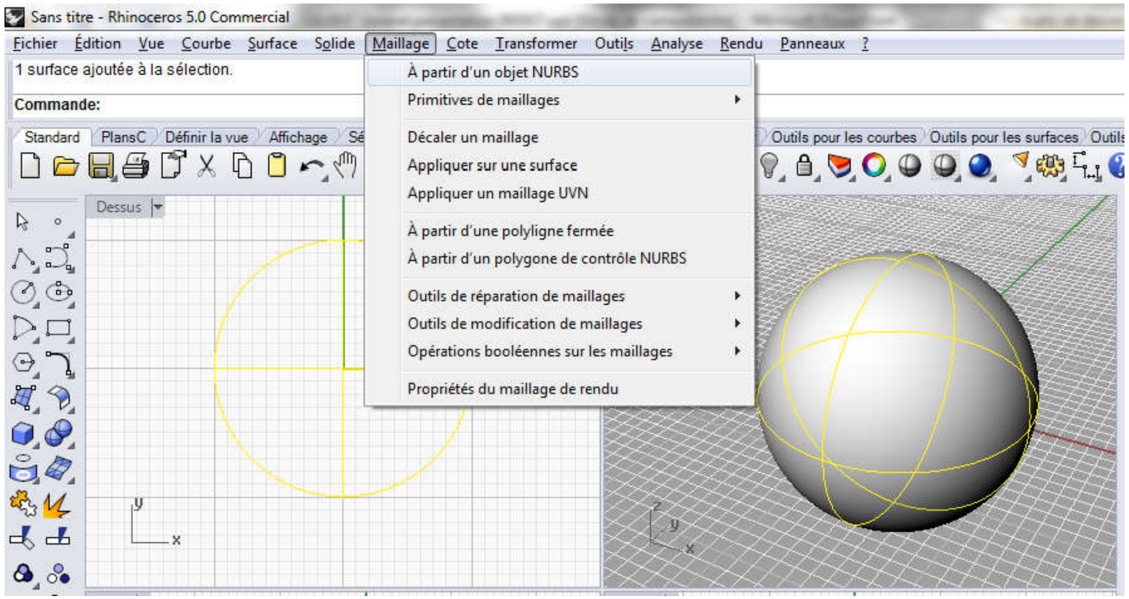
-Option 2 : Import geometry for other CAD software : many file formats can be imported

(IGES, STEP are nice solutions).

-Option 3 : Import .stl file from any source.

Prefer other options, as you won't be able to adjust the size of the .inp file.

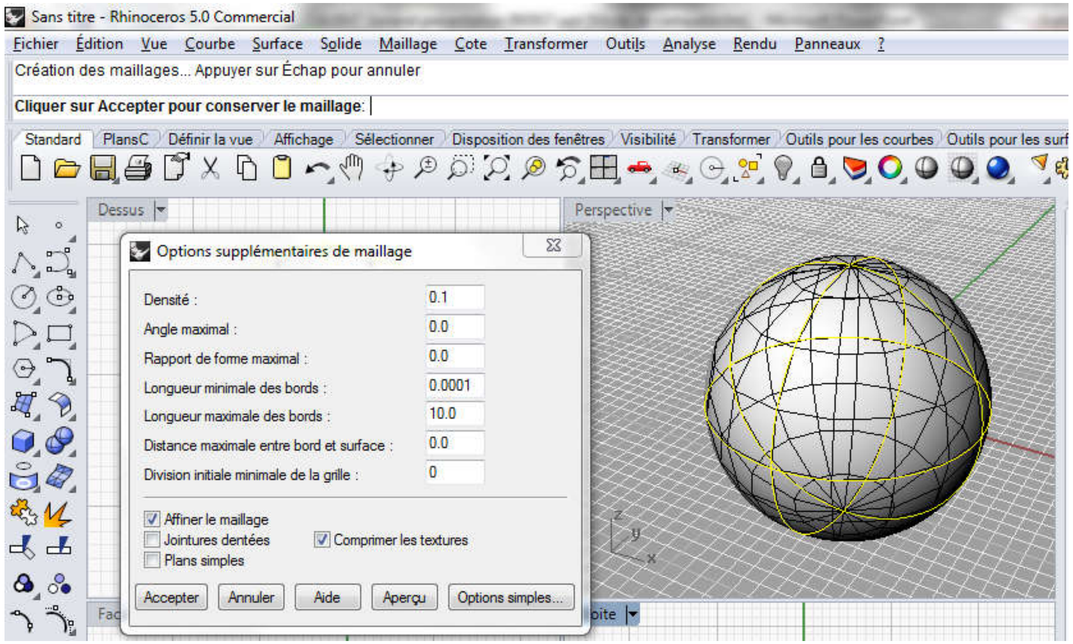
STEP 3 : Preparing the meshes in rhino5



Use this function to create a mesh for each object to be imported into MCNP

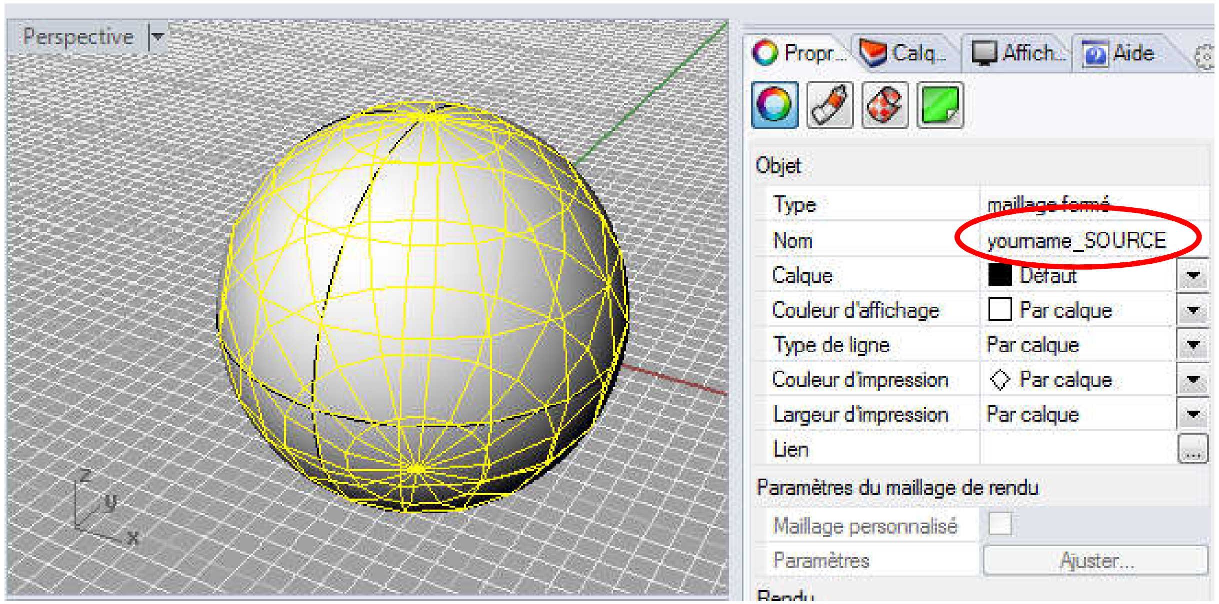
SKIP STEP 3 when using .STL file

STEP 3 : Preparing the meshes



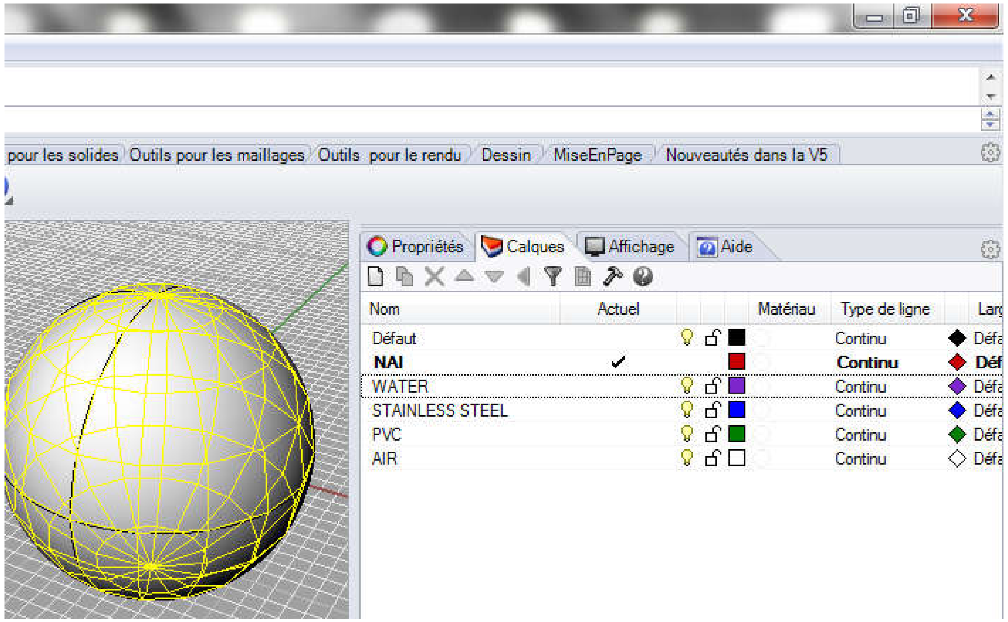
Advised starting meshing parameter.

STEP 4 : Rename the meshes



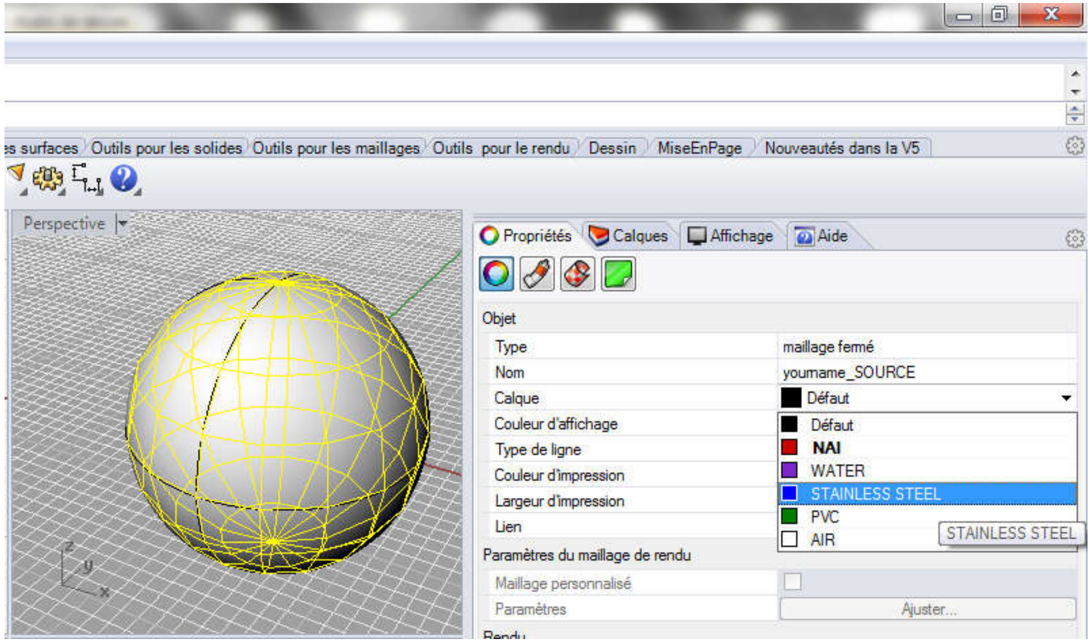
Keywords « source » and « detector » can be used to designate source and detector objects. Several detectors can be carried into MCNP. Only one source can be declared (limitation of MCNP6.11)

STEP 5 : Create layer set for materials



Rename a layer for each material you want to declare into MCNP

STEP 6 : Assign all meshes to their layers



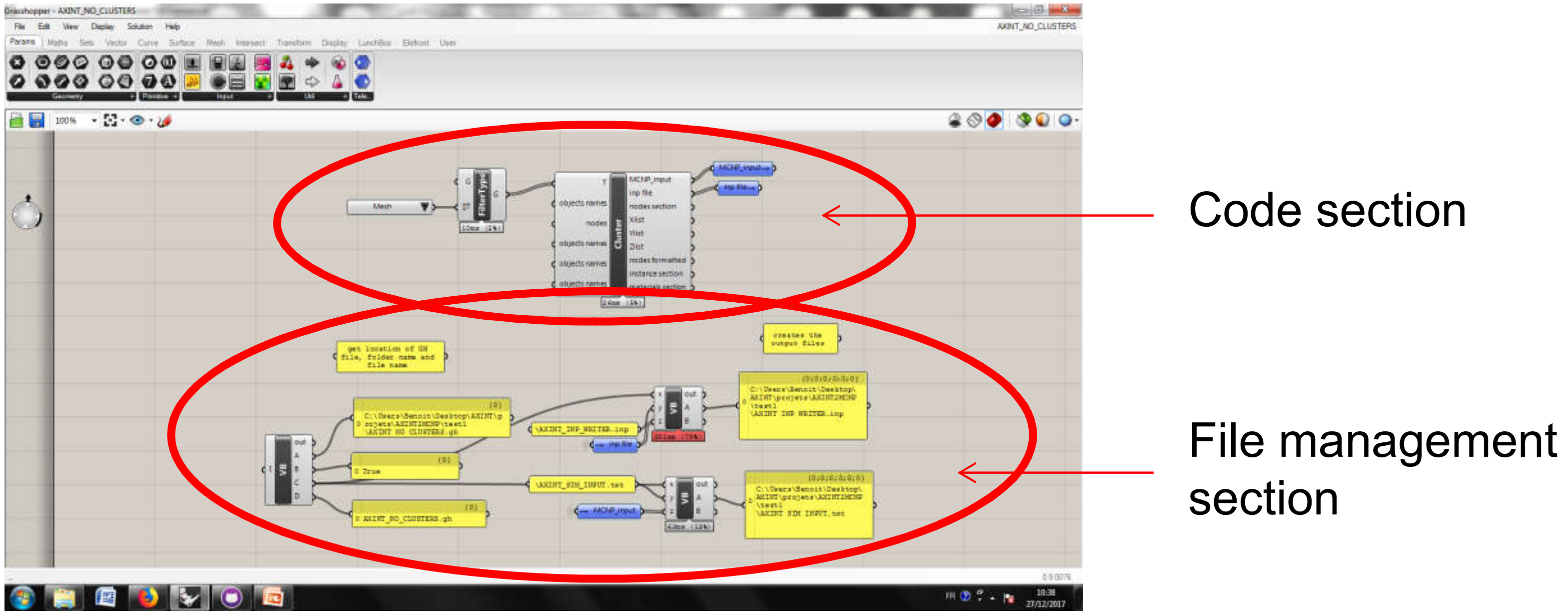
Several meshes can be be assigned with the same material

STEP 7 : Save your Rhino file.

PROJECT IS READY FOR CONVERSION !

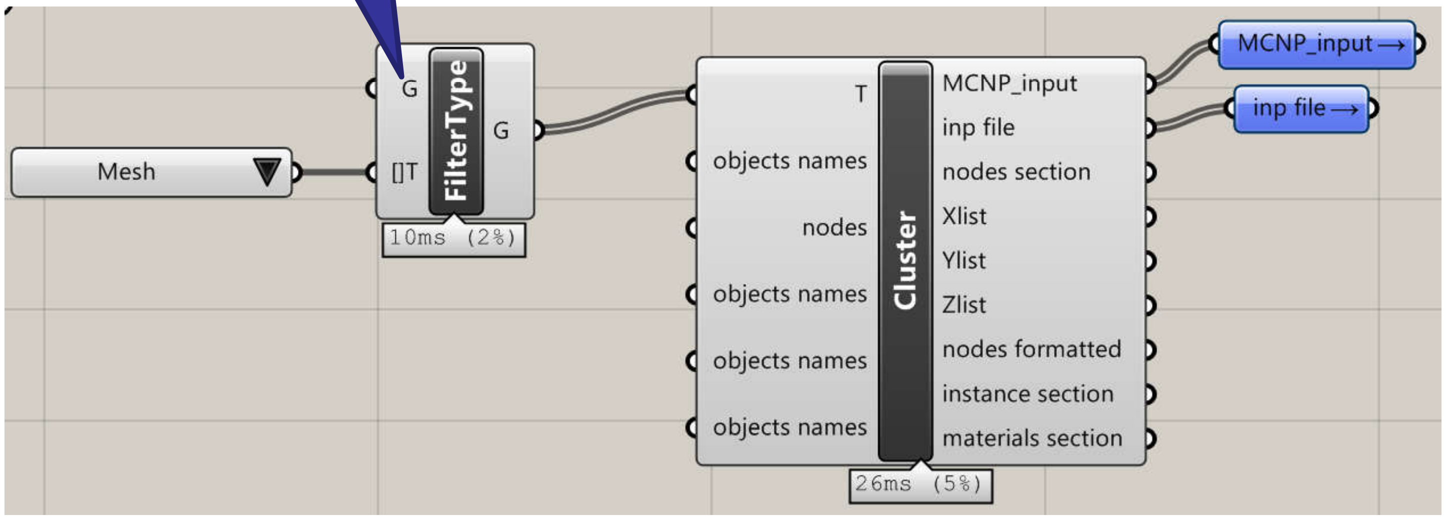
File conversion step 1 :

-In Rhinoceros 5, type the grasshopper command. The open the « AXINT-INP_WRITER.gh » file;



Right click on « G » Then « set multiple extended geometry ». Select your meshes in rhino.

In the code section :



Note : you can use CTRL+A to select all objects in Rhino. Any non-mesh object will be ingnored by the converter.

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Step 4

Conversion will start automatically.

Be carefull, large file can take some time for files conversion. It is advised to verify the number of facets in the Rhino file wil the function Analysis → Diagnostics → List. An object with 10K+ facets may take more than 1 minute for conversion.

Another way to estimate the conversion time is the size of the Rhino file : A Rhino file size of 3Mo will take about 1 minute for conversion (depending on computer…)

Step 5 : inspect AXINT_SIM_INPUT .txt

This is the main MCNP input file. The prorotype file is automatically created, but you need to add some more information by hand. The prototype file includes comments to guide the user for simple simulations.

Users may refer to MCNP6 manual for more complex simulations.

Step 5 : inspect AXINT_SIM_INPUT .txt

```
WRITTEN BY AXINT INP WRITER
101 1 'insert (-)density for aluminium'    0 u=1 imp:n=1   $   object name : crystal can      tetrahedrons : 10440
102 1 'insert (-)density for aluminium'    0 u=1 imp:n=1   $   object name : source      tetrahedrons : 511
103 1 'insert (-)density for aluminium'    0 u=1 imp:n=1   $   object name : housing      tetrahedrons : 19641
104 1 'insert (-)density for aluminium'    0 u=1 imp:n=1   $   object name : clamp      tetrahedrons : 4399
105 2 'insert (-)density for PA12'    0 u=1 imp:n=1   $   object name : 3D printed 1      tetrahedrons : 29215
106 3 'insert (-)density for BK7'    0 u=1 imp:n=1   $   object name : window      tetrahedrons : 438
107 4 'insert (-)density for NaI'    0 u=1 imp:n=1   $   object name : detector      tetrahedrons : 523
108 5 'insert (-)density for FR4'    0 u=1 imp:n=1   $   object name : PCB1      tetrahedrons : 2686
109 5 'insert (-)density for FR4'    0 u=1 imp:n=1   $   object name : PCB2      tetrahedrons : 2761
110 5 'insert (-)density for FR4'    0 u=1 imp:n=1   $   object name : PCB3      tetrahedrons : 67354
111 2 'insert (-)density for PA12'    0 u=1 imp:n=1   $   object name : 3D printed 2      tetrahedrons : 43369
112 2 'insert (-)density for PA12'    0 u=1 imp:n=1   $   object name : 3D printed 3      tetrahedrons : 7963
113 'insert material index for background' 'insert (-)density for background' 0 u=1 imp:n=1   $    background cell
114 0 -99 fill=1 imp:n=1   $    fill cell
115 0 99 -98 imp:n=1
116 0 98 imp:n=0
C report : total number of tetrahedrons in the project :  189300
```

Cells section : the user may enter here the density of each object. Names are reported for easy use (1), as well as tetrahedrons count for each object (2) and total count in the project (3).

Step 5 : inspect AXINT_SIM_INPUT .txt

```
WRITTEN BY AXINT_INP_WRITER
101 1 'insert(-)density for aluminium'    0 u=1 imp:n=1   $   object name : crystal can      tetrahedrons : 10440
102 1 'insert(-)density for aluminium'    0 u=1 imp:n=1   $   object name : source      tetrahedrons : 511
103 1 'insert(-)density for aluminium'    0 u=1 imp:n=1   $   object name : housing      tetrahedrons : 19641
104 1 'insert(-)density for aluminium'    0 u=1 imp:n=1   $   object name : clamp      tetrahedrons : 4399
105 2 'insert(-)density for PA12'    0 u=1 imp:n=1   $   object name : 3D printed 1      tetrahedrons : 29215
106 3 'insert(-)density for BK7'    0 u=1 imp:n=1   $   object name : window      tetrahedrons : 438
107 4 'insert(-)density for NaI'    0 u=1 imp:n=1   $   object name : detector      tetrahedrons : 523
108 5 'insert(-)density for FR4'    0 u=1 imp:n=1   $   object name : PCB1      tetrahedrons : 2686
109 5 'insert(-)density for FR4'    0 u=1 imp:n=1   $   object name : PCB2      tetrahedrons : 2761
110 5 'insert(-)density for FR4'    0 u=1 imp:n=1   $   object name : PCB3      tetrahedrons : 67354
111 2 'insert(-)density for PA12'    0 u=1 imp:n=1   $   object name : 3D printed 2      tetrahedrons : 43369
112 2 'insert(-)density for PA12'    0 u=1 imp:n=1   $   object name : 3D printed 3      tetrahedrons : 7963
113 'insert material index for background' 'insert(-)density for background' 0 u=1 imp:n=1   $    background cell
114 0 -99 fill=1 imp:n=1   $    fill cell
115 0 99 -98 imp:n=1
116 0 98 imp:n=0
C report : total number of tetrahedrons in the project :  189300
```

Background material : In the MCNP6 Unstructured mesh feature, the background material is the material that will fill the space non described by the meshes. (This is often air). User may insert material index and density for background. This can be one of the already listed materials, or a new one.

Step 5 : inspect AXINT_SIM_INPUT .txt

```
C SURFACE CARDS
99 RPP -49.25 49.25 -49.25 49.25 -206 36    $   fill cell
C UNIVERSE
98 RPP -50.25 50.25 -50.25 50.25 -207 37

C ABAQUS/CAE EMBEDDED MESH SECTION
embed1 meshgeo=abaqus mgeoin=AXINT_INP_WRITER.inp &
meeout=leadwall_abq3.eeout background=113 filetype=ascii &
matcell= 1 101 2 102 3 103 &
4 104 5 105 6 106
7 107 8 108 9 109
10 110 11 111 12 112
embee4:p embed=1            $Elemental tally
```

This section should does not require intervention from the user.

(1) is a bounding box including all meshes, automatically generated by the converter.

(2) is the declaration of the MCNP cell corresponding to the list of meshes.

Step 5 : inspect MCNP INPUT FILE

```
m1 'add here aluminium material definition'
m2 'add here PA12 material definition'
m3 'add here BK7 material definition'
m4 'add here NaI material definition'
m5 'add here FR4 material definition'
mode p     $  (p=photons, e=electrons, n=neutrons)
```

Materials definition simplified guideline (refer to MCNP manual for more options) :

Materials definition should include atomic number and atomic mass (or 000 for natural isotopic composition), and atomic fraction (not necessary normalized) for each element in the material

Ex1 : Water : Z(H) = 1, atomic fraction = 2, Z(O) = 8, atomic fraction = 1, natural isotopic composition (000)

Material corresponding definition : m1 1000 2 8000 1

Ex2 : 235U : Z(U) = 92, atomic fraction = 1 atomic masss (235)

Material corresponding definition : m1 92235 1

Step 5 : inspect MCNP INPUT FILE

```
mode p     $  (p=photons, e=electrons, n=neutrons)
SDEF ERG = 'XXX(MeV)'or'D1' POS = volumer     $ indicate source energy or use D1 if source is not monoenergetic
SI1 L 0.511 1.275                             $ list of energies. remove line if D1 is not used
SP1 1.8 1                                     $ relative probabilities. remove line if D1 is not used
f18:p 107
E18 a ni b                  $ for energy histogram results a=energymin b=energymax n=channels count
FT18 GEB 0.01 0.08 0        $ for gaussian flattening of the tally results (i.e. scintillator energy resolution)
nps 5000000                 $ number of particles to be simulated
```

-Mode : « p » means that photons will be transported. Change to «e» to transport electrons, « n » for neutrons. « p e » for photons + electron, aso… Default value is p, user may edit this line.

-SDEF line : source declaration : replace XXX by the source enegy for monoenergetic source, or type « D1 » if multiple energies emitted. In this case, list the energies in SI1 line, and probabilities in SP1 line. Default value descibe a well known 60Co source as an example.

Remark : source will be emitted in the mesh with the « source » keyword. Multiple mesh sources are not supported currently, due to a MCNP6.11 limitation. It will be developped when MCNP will be able to handle it.

Step 5 : inspect MCNP INPUT FILE

```
mode p     $  (p=photons, e=electrons, n=neutrons)
SDEF ERG = 'XXX(MeV)'or'D1' POS = volumer      $ indicate source energy or use D1 if source is not monoenergetic
SI1 L 0.511 1.275                              $ list of energies. remove line if D1 is not used
SP1 1.8 1                                      $ relative probabilities. remove line if D1 is not used
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FT18 GEB 0.01 0.08 0        $ for gaussian flattening of the tally results (i.e. scintillator energy resolution)
nps 5000000                 $ number of particles to be simulated
```

f18 line : tally of the simulation : default value is an F18 tally (the simulation will return the number of interaction in the target cell (indentified by the « detector » keyword). Use may want to edit this line.

E18 line : Used to simulate a scintillation detector : a and b describe the energy range, and n is the channel number. The simulation will return an energy histogram as a result :

FT18 line : gaussian flattening : used to simulated the energy resolution for a scintillation detector. Refer to MCNP6 manual for more details.

NPS : number of particules to be emitted.

Step 5 : inspect MCNP INPUT FILE

```
mode p      $  (p=photons, e=electrons, n=neutrons)
SDEF ERG = 'XXX(MeV)'or'D1' POS = volumer      $ indicate source energy or use D1 if source is not monoenergetic
SI1 L 0.511 1.275                              $ list of energies. remove line if D1 is not used
SP1 1.8 1                                      $ relative probabilities. remove line if D1 is not used
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nps 5000000                  $ number of particles to be simulated
```

NPS : number of particules to be emitted. User may edit this line.

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Step 6 : Run the simualtion

To run the simulation, the AXINT_SIM_INPUT.txt and the AXINT_INP_WRITER.inp files must be located in the same folder. AXINT_INP_WRITER.inp file includes all the meshes information, it can be a heavy file. There is no intervention required on this file.

For more information, contact hautefeuille@axint.fr

Remark :

It is possible to include a list of usual material so that they can be included automatically in the simulation input file. This is not implemented in the code yet, I will do it if we decide to ditribute this program to MCNP beginners.