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Title: Criticality Accident Alarm System (CAAS) CSG-UM Hybrid Example

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Intended for: MCNP6 Training: Unstructured Mesh with Attila4MC
Web

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Criticality Accident Alarm System (CAAS)
CSG-UM Hybrid Example

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CAAS Example

Criticality Accident Alarm System calculations combine methods needed for both criticality and shielding calculations. UM may be used for facility details, allowing import of existing facility drawing. CSG used for criticality cells.

Objectives:

- Demonstrate a hybrid method using unstructured mesh for facility and CSG for criticality cells
- Import solid geometry, generate mesh, create calculation in Attila4MC and pack for MCNP
- Generate a source file from KCODE calculation in MCNP
- Define tallies to calculate energy deposition to a detector
- Employ variance reduction techniques to obtain statistically significant results

CAAS Example

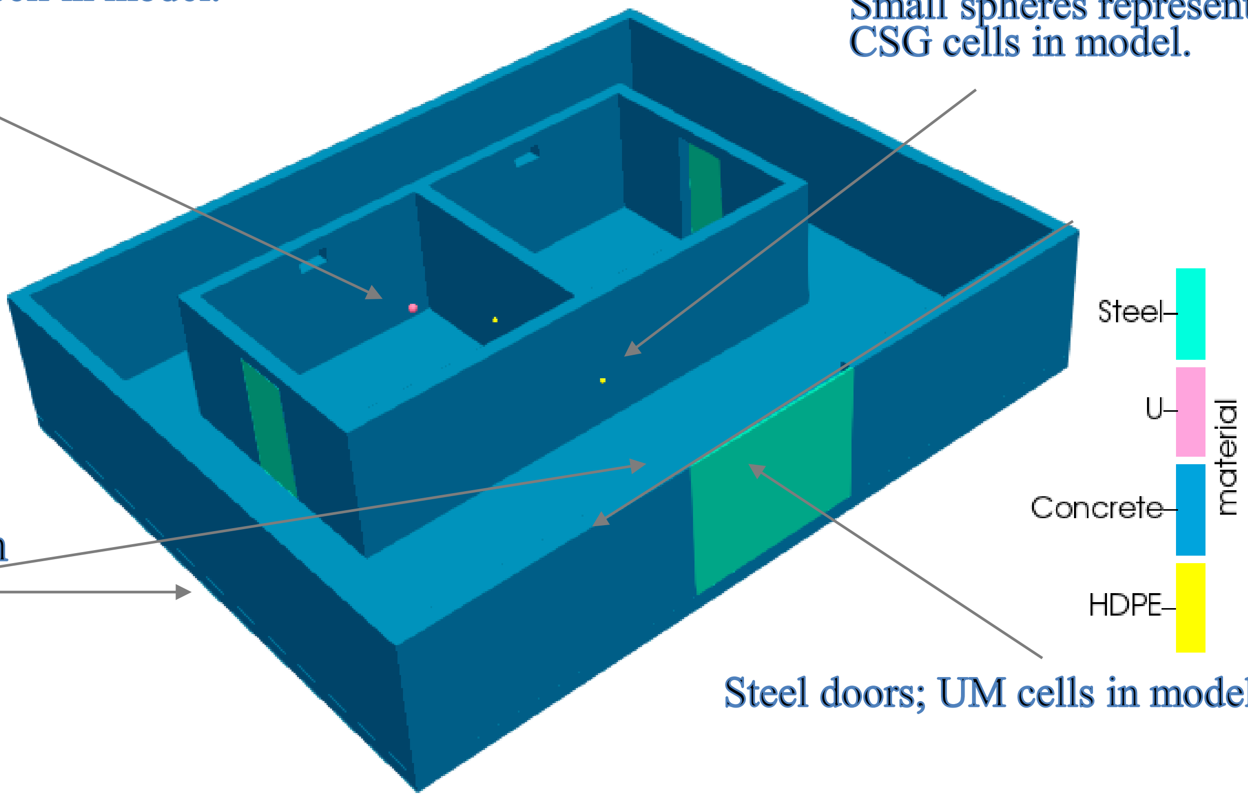
Steps in Hybrid Method:

- Solid geometry import into Attila4MC
- Generate Mesh in Attila4MC
- Create calculation in Attila4MC and pack for MCNP
- Modify MCNP6.2 input file for insertion of CSG cells
- Run MCNP6.2 KCODE calculation with SSW
- Run MCNP6.2 fixed source calculation with SSR
- Variance Reduction steps to obtain detector result

CAAS Example

HEU sphere; CSG cell in model.

Small spheres representing detectors; CSG cells in model.



Concrete walls, floor; UM cells in model.

Steel doors; UM cells in model.

CAAS Example

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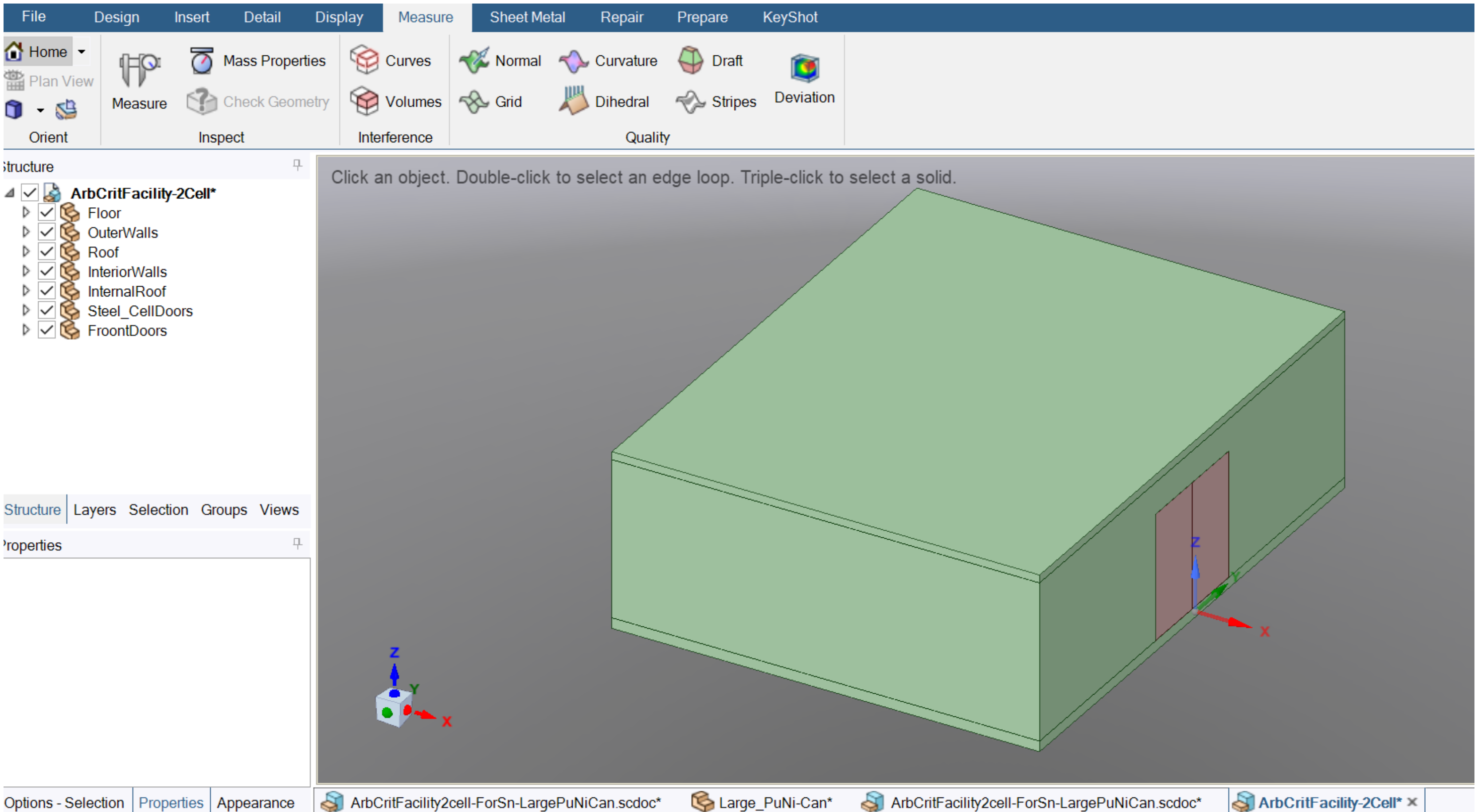
CAAS Example

Solid geometry import into Attila4MC

ArbCritFacility-2Cell.x_t -- view parasolid file in Spaceclaim → import into Attila4MC



CAAS Example

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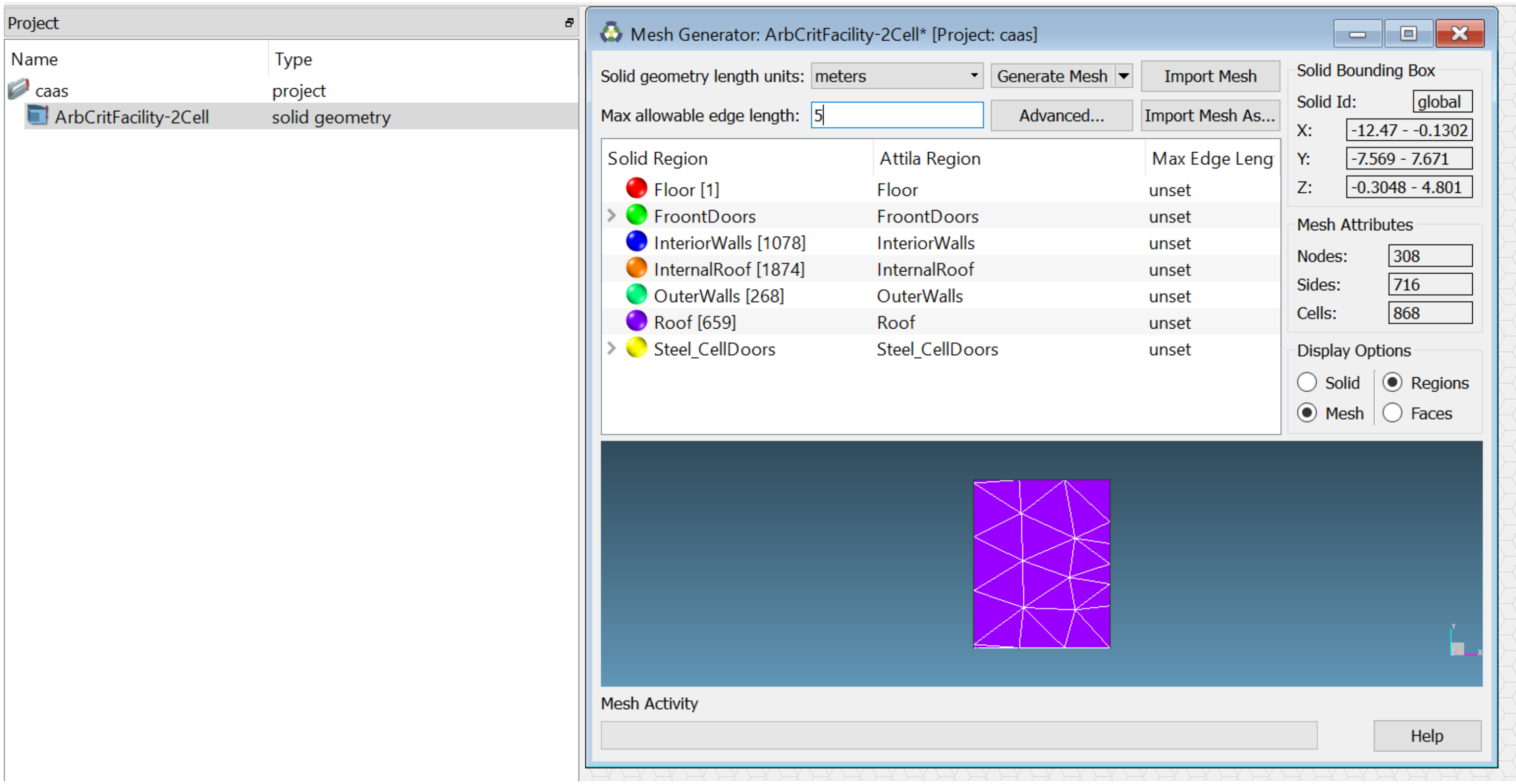
CAAS Example

Generate Mesh in Attila4MC:

File: New: Project: caas

Project: Import file: Solid Geometry: **ArbCritFacility-2Cell.x_t**

Generate Mesh (unclick Identify Empty Regions) → **ArbCritFacility-2cell.mesh.inp**



CAAS Example

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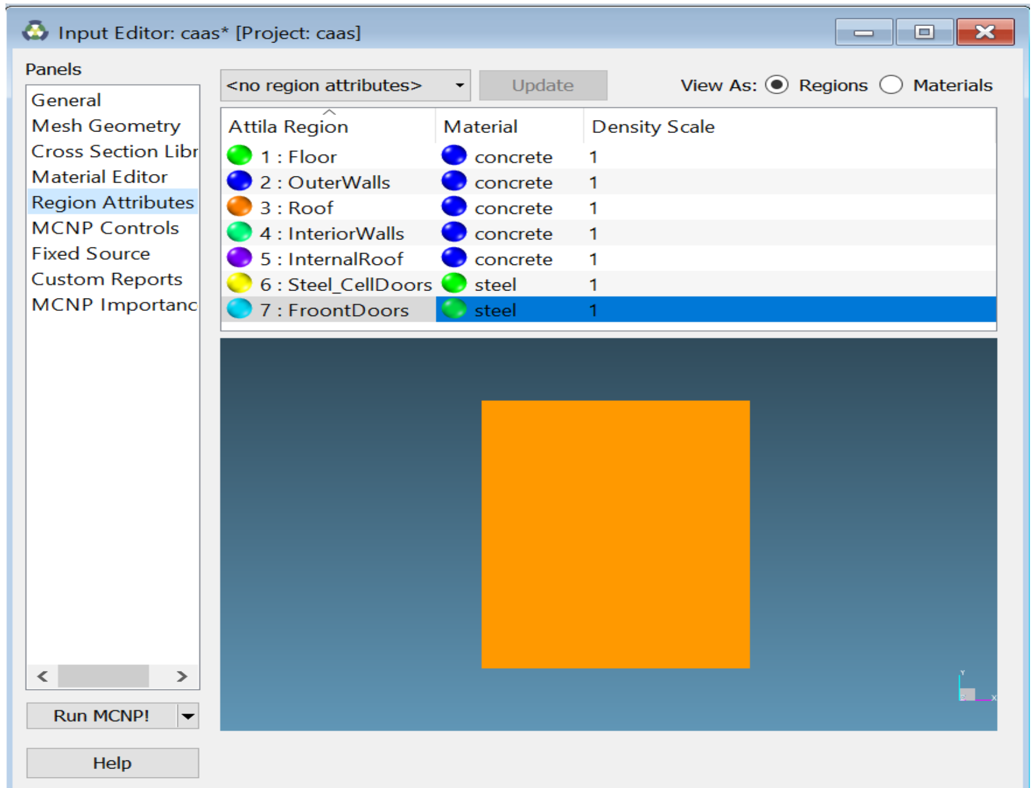
CAAS Example

Create calculation in Attila4MC and pack for MCNP

Project: Create Object > Calculation
Library Name: **MCNP_Isotopes_Lib.xs.aux.inp** (must have cross sections imported)
Material – choose material for each Attila region
To import materials – Project: Import MCNP Attribues…**arb_crit_facility-CSG.mcnp**.i

Use panels to specify

Calculation> Pack for MCNP
arbcritfacility-2cell.mcnp.i
arbcritfacility-2cell.abaq



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CAAS Example

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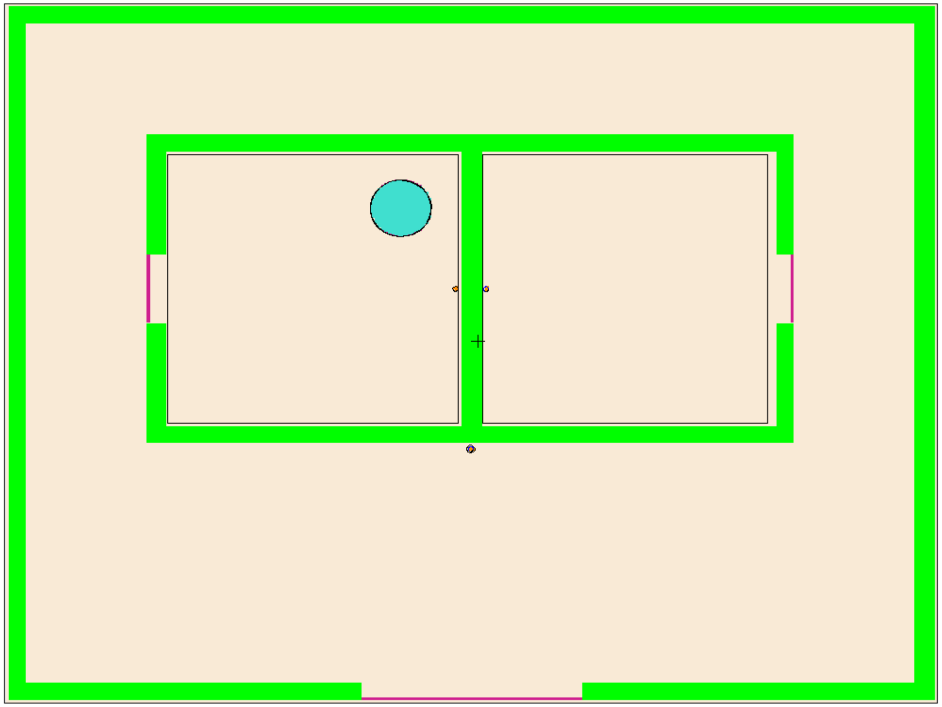
CAAS Example

Modify MCNP6.2 input file for insertion of CSG cells



```
1    2    -2.3          0           u=1
2    2    -2.3          0           u=1
3    2    -2.3          0           u=1
4    2    -2.3          0           u=1
5    2    -2.3          0           u=1
6    1    -7.6          0           u=1
7    1    -7.6          0           u=1
8    3    -1.3e-3       0           u=1
9    3    -1.3e-3     -100 201 202 203 fill=1
21   3    -1.3e-3                   -201 #30 #40
22   3    -1.3e-3                   -202 #41
23   3    -1.3e-3                   -203 #42
30   92       -18.74    -301
40   4  -0.96                       -401
41   4  -0.96                       -402
42   4  -0.96                       -403
99   0                  100
```

CAAS Example

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CAAS Example

- **MCNP6.2 KCODE calculation with SSW**
- **Generated with the SSW or "Surface Source Write" card.**
- **Form: SSW CEL = C1 C2 …**
  - The arguments after the CEL keyword is a list of cells to store fission source points for in a KCODE calculation.
  - Produces a file with default name wssa.
  - Additional options are available. See the MCNP Manual.

Surface Sources

- **MCNP can generate a binary file called a “surface source” file.**
  - Normally, this contains particle tracks that crossed a surface, which are run in a different calculation
- **The “surface source” file may also contain fission source points from a KCODE calculation.**
- **The surface source file may be used as the source from a criticality accident.**
- **Note: Surface sources do not yet work with OMP threading.**

Surface Source Considerations

- **Some considerations when deciding how many fission points to bank:**
  - Define NSS as the number of particles banked.
  - A sufficient number of fission source points is needed describe a continuous neutron field – more is usually better.
  - When reading the file, MCNP allows a variable number of particles NPS to be used.
  - If NPS < NSS, then NPS particles are randomly selected with an increased starting weight per history.
  - If NPS > NSS, then NSS starting particles are started, but some will be duplicated randomly with lower starting weight per history. Note that the NPS used for normalizing tallies is the same.

Generating a Surface Source

```
…
KCODE  20000  1.0   50  150
KSRC   -888.38    -110.15    108.5
ssw cel=30
…
```

- KCODE card to use 20,000 neutrons per cycle, initial guess for k-effective 1.0, skip 50 cycles, and run 100 active cycles
- Specify one source point in the center of the plutonium nitrate
- Add SSW card for cell 30.
- Run the problem, name the wssa file source.

```
mcnp6  i = caas1.txt  o = caas1o.txt wssa = source
```

- Note: This will generate 2 million source points and will take a long time.

- Examine results, verify run was successful and Shannon entropy check confirms source convergence

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Surface Source Read

- **May read the surface source in a fixed source calculation.**
- **Form: SSR CEL = C1 C2 … WGT = W PSC = P**
  - The arguments after the CEL keyword is a list of cells to use from the surface source file.
  - W is the intensity of the source (neutrons released from the burst).
  - PSC is the probability of scattering cosine. From fission this is isotropic and 0.5. This is needed for F5 tallies and DXTRAN.
  - Reads a file with default name rssa.
  - Additional options are available. See the MCNP Manual.

Neutron Fission Treatment

- **Since fission was already treated in the KCODE calculation, it must be treated as capture.**

- **Form: NONU N1 N2 … N(NCEL)**

  - Specifies a list of cells where fission is treated as capture
    = 0, do not perform fission (treat as capture)
    = 1, perform fission
  - Must list “0” for each cell in the problem. May also do this on the cell card.
  - If this is not done, the problem will run forever because it is supercritical. Even if it were subcritical, the answer would be wrong.

Energy Deposition (F6) Tally

- **The neutron energy deposited in a cell may be obtained with an F6 tally.**

- **Form: F6:n C1 C2 …**

  - Computes energy deposition for each cell listed on the card in MeV/gram.
  - Otherwise, very much like an F4 tally, i.e., may use FM cards, etc.

Reading the Surface Source

- **Copy caas1.txt to caas2.txt.**
  - Modify the eeout file to meeout=caas2.mcnp.eeout
  - Delete the SSW card and insert a SSR card.
  - The intensity of the burst is 1e15 fissions times 2.9 neutrons per fission.
  - Delete KCODE and insert an NPS card with 1e5 neutrons.
  - Insert a NONU card and turn off fission in all cells.
  - Create an energy deposition tally for cell 40. Convert the units of the tally to Gy or J/kg (1.602e-10 is the conversion factor from MeV/gram to J/kg).
  - Run the problem reading the file source.

```
mcnp6  i = caas3.txt  o = caas3o.txt  rssa = source
```

  - Analyze the tally output.

Caas1.txt

```
. . . . .
kcode  20000   1.0   50  150
KSRC   -888.38 -110.15 108.5
ssw cel=30
. . . . .
```

Caas2.txt

```
. . . . .
c Source Definition
ssr cel=30 wgt=2.9e15 psc=0.5
nonu 0 18r
c
c Histories (or Computer Time Cutoff)
nps 1e6
c
c Tallies or embee cards
fmesh4:n ORIGIN=-1255. -750. 0.
        IMESH=0.  IINTS=184
        JMESH=750.  JINTS=124
        KMESH=450.  KINTS=36
c
f16:n 40
fm16 1.6022e-10
fc16 Criticality Accident Dose at Detector 1 in Gy
```

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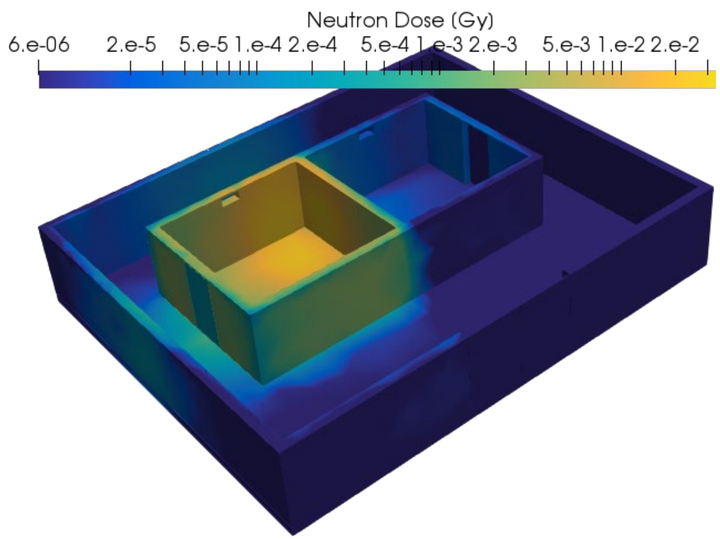
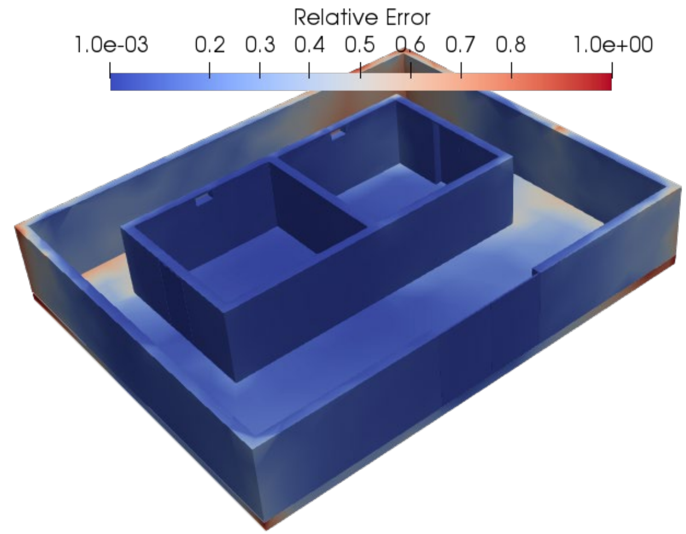
CAAS Example

- Results using FW-CADIS weight windows compared with analog





Geometry	Detector 1 Dose in Gy (RE)	FOM > Analog	Detector 2 Dose in Gy (RE)	FOM > Analog	Detector 3 Dose in Gy (RE)	FOM > Analog
CSG	$1.48\ 10^{-1}$ (0.01)	4X	$7.34\ 10^{-4}$ (0.01)	7879X	$1.47\ 10^{-4}$ (0.06)	676X
CSG/UM	$1.47\ 10^{-1}$ (0.01)	4X	$7.42\ 10^{-4}$ (0.01)	583X	$1.47\ 10^{-4}$ (0.03)	661X
UM	$1.47\ 10^{-1}$ (0.01)	2X	$7.42\ 10^{-4}$ (0.01)	550X	$1.47\ 10^{-4}$ (0.03)	196X