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Title: Calculations of radiation back-flux from neutron irradiation in fusion reactors

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Calculations of radiation back-flux from neutron irradiation in fusion reactors

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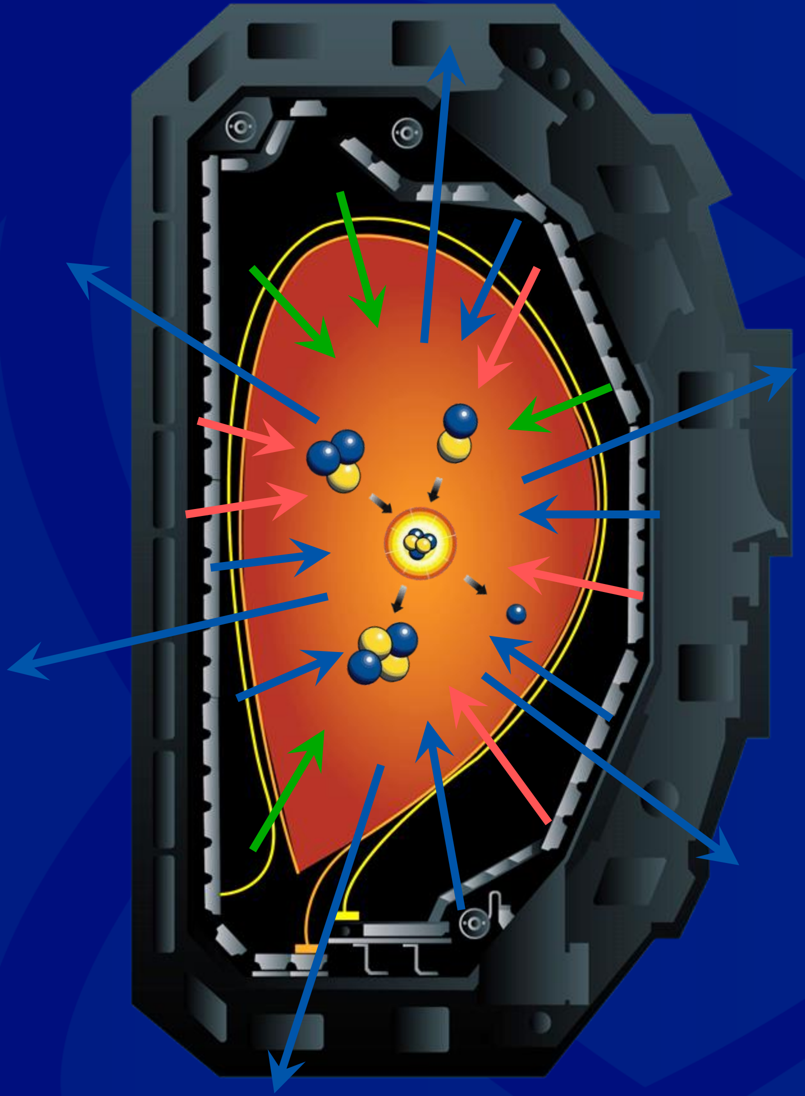
2024 MCNP® User Symposium
Wed 21 August 2024, 11:10-11:30



Nuclear fusion in 60 seconds

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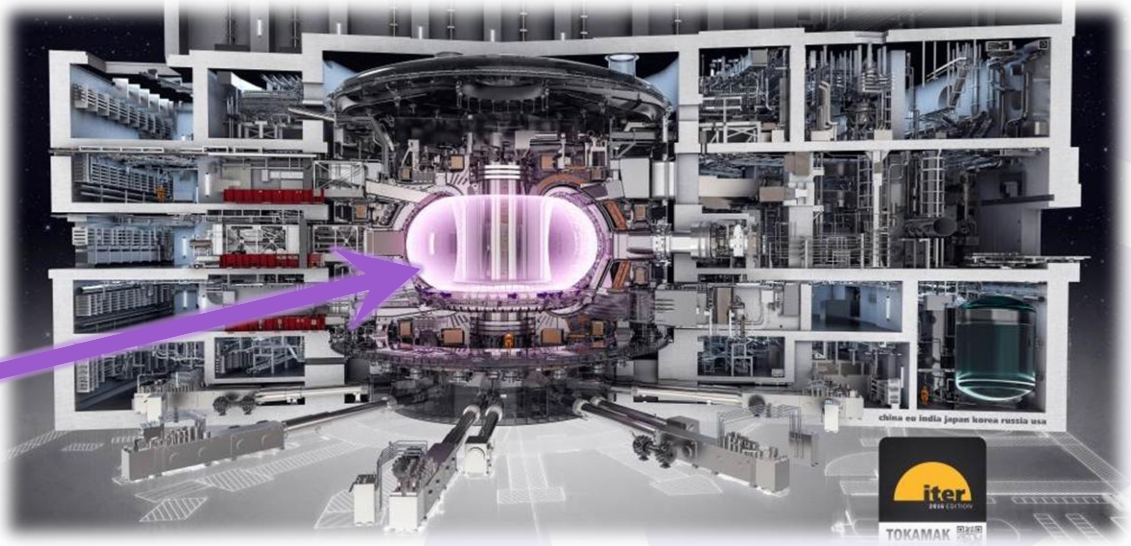
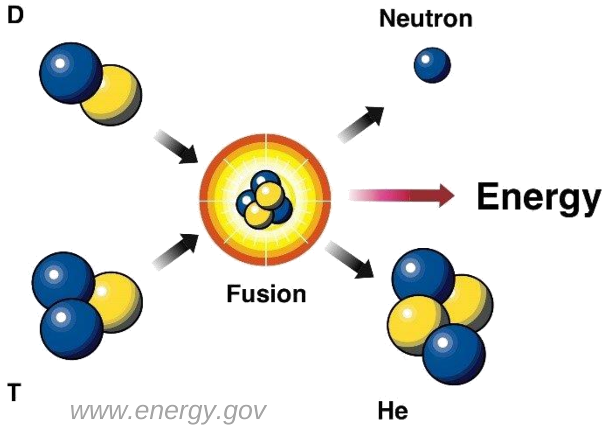


Nuclear fusion takes place inside a plasma, a super-hot gas (100,000,000° C) of ions and electrons.



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When the plasma gets hot enough, smaller ions can collide and form a bigger ion, releasing astronomical amounts of energy.

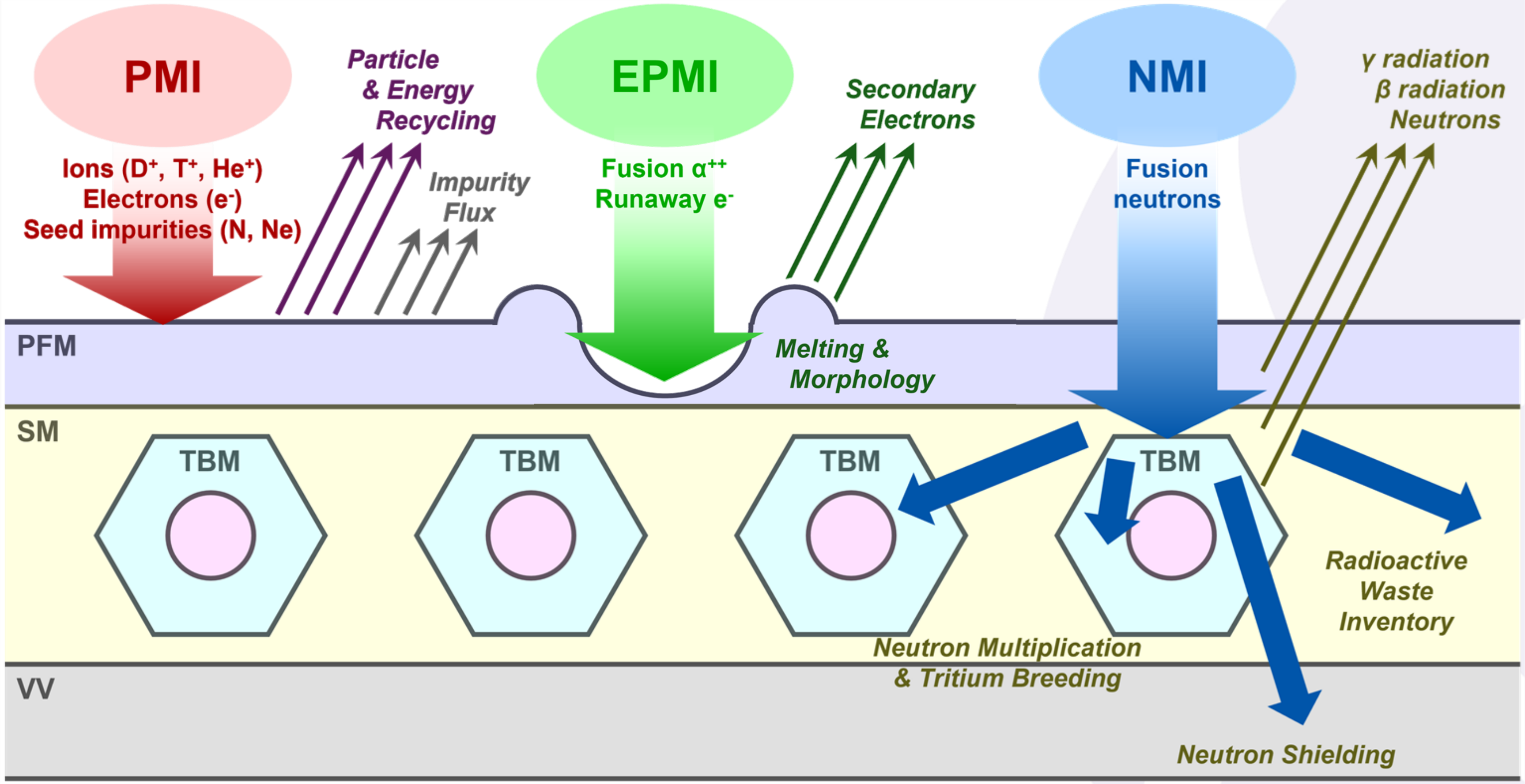
How much energy?

1 gal water → 10.7 GJ

~16 gallons of water

8 tons of coal!

Plasma-material interactions in a fusion power reactor

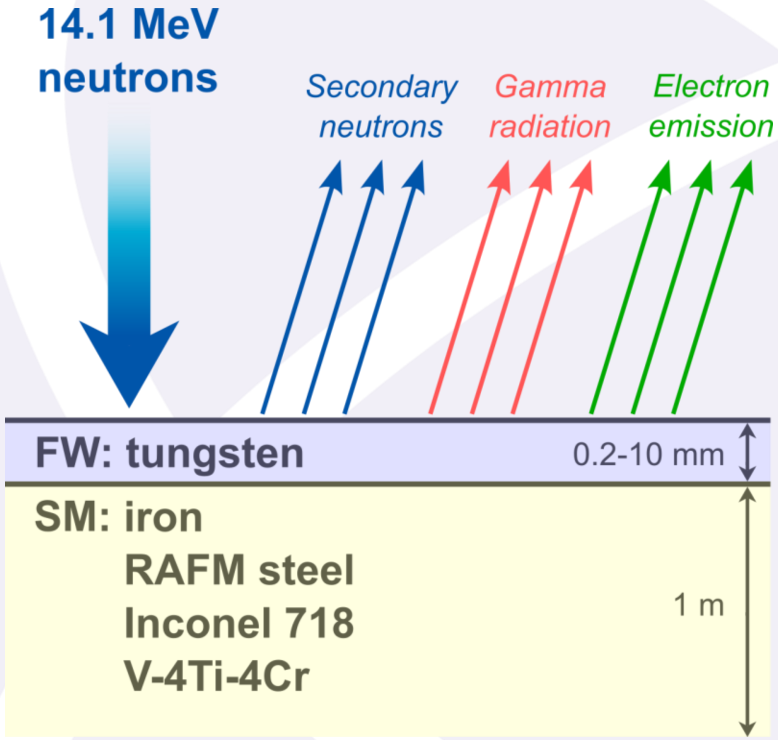


In a burning plasma producing high-energy (14.1 MeV) fusion neutrons, neutron-material interactions (NMI) drive new plasma physics processes.

We require models spanning a diverse range of radiation types (neutrons, gamma rays, electrons, ions) in complex fusion reactor geometries.

First-step MCNP simulations towards understanding neutron irradiation induced back-flux from the wall

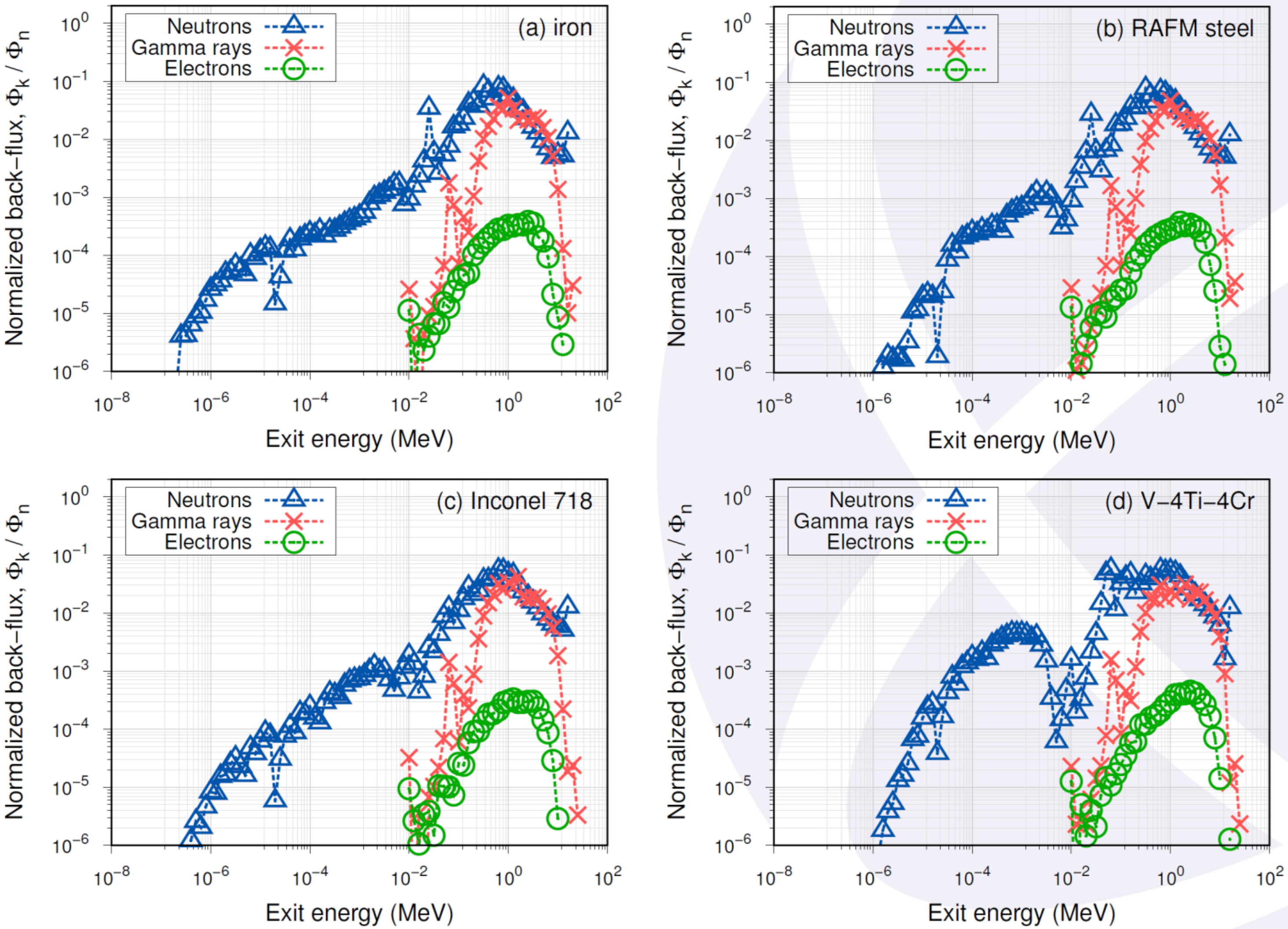
- We will compare two calculation approaches:
  - Explicit approach (**`FT8 RES`** card + decay solver)
    - Nuclide production tally → decay solver for $\{N(t)\}$ → gamma/beta emission distributions → MCNP source distribution
    - MCNP calculations done in this manner are relatively quick, residual nuclide information is useful for ancillary studies.
  - Implicit approach (**`ACT`** card)
    - Does all of the above in one step within MCNP ("black box"), therefore a simpler approach in principle.
    - However, simulations take much longer (~20-25x) and we don't get the ancillary nuclide inventory data.
  - Our goal is to compare both approaches for the benefit of the fusion community.
    - Implicit solution acts as verification for explicit solution.
- Simulation parameter space (at right):
  - 4 structural materials (SM) – subdivided into 20 cells for **`F8`** tally
  - 4 first wall (FW) thicknesses
  - 4 × 4 = <u>16 total simulations</u>



MCNP calculation setup and materials design parameters for this work.

Energy-resolved radiation back-fluxes from 2-mm tungsten first wall with various structural materials

<u>Iron</u>:
$\phi_{n'} = 0.8756$
$\phi_{\gamma} = 0.3587$
$\phi_{e} = 0.0042$

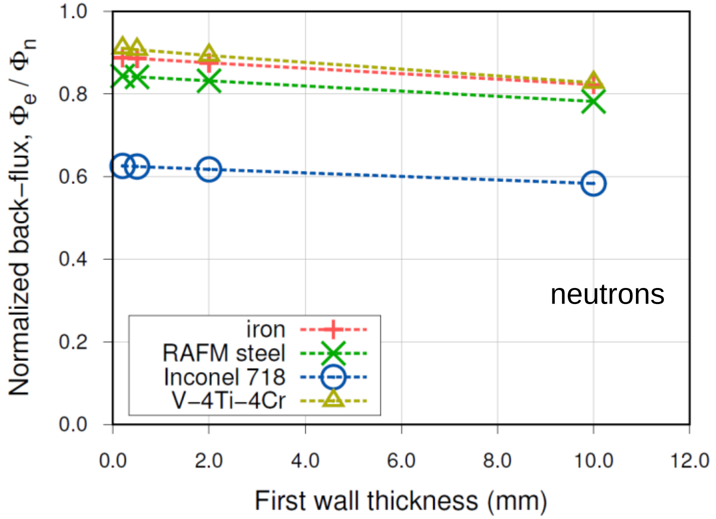


<u>RAFM steel</u>:
$\phi_{n'} = 0.8321$
$\phi_{\gamma} = 0.3534$
$\phi_{e} = 0.0041$

<u>Inconel 718</u>:
$\phi_{n'} = 0.6178$
$\phi_{\gamma} = 0.3158$
$\phi_{e} = 0.0037$

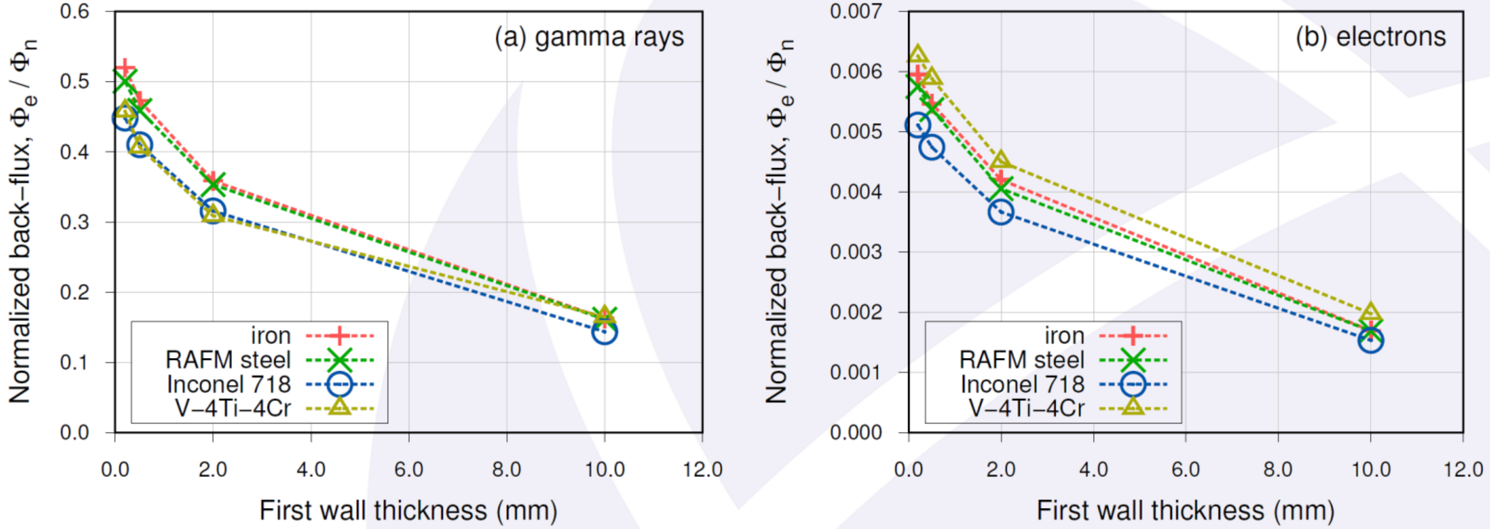
<u>V-4Ti-4Cr</u>:
$\phi_{n'} = 0.8931$
$\phi_{\gamma} = 0.3089$
$\phi_{e} = 0.0045$

Trends in total radiation back-fluxes with varied FW thickness and SM selection



<u>Trends for neutron back-flux</u>:

- Weak dependence on FW thickness.
- Strong dependence on SM choice; Inconel 718 emits 25-30% lower neutron back-fluxes.



<u>Trends for gamma radiation and electron emission back-fluxes</u>:

- Strong dependence on FW thickness indicates effective attenuation by tungsten (i.e., large stopping power in high-Z materials).
- Weak dependence on SM choice, although Inconel 718 generally emits the lowest back-flux levels.
- Electron back-flux is consistently ~2 orders of magnitude lower than gamma radiation back-flux.

Implications of radiation back-fluxes for fusion power reactor operations and plasma performance

- Impact on reactor power balance:
  - On the order of ~10% of incident fusion neutron power (~8% of total fusion power) is radiated back to the plasma.
  - Of this, ~60% is carried by the neutron back-flux, which is unlikely to interact with the plasma.
    - Redeposition at other plasma-facing (or not?) surfaces.
  - ~40% is carried by the gamma radiation back-flux, which may interact with the plasma by, e.g., Compton scattering.
    - Or redeposit elsewhere - σ_c is not large, so λ_{mfp} is quite long ($\sim 10^7$ m).
  - ~0.4% is carried by electron back-flux, which will be magnetically trapped near the plasma edge
    - Large localized heating – could be dangerous/unstable.
- Impact of neutron multiplication:
  - Back-flux magnitudes reported here for *gamma rays* and *electrons* increase by order of 50-80%.
    - This is calculated by a reflecting boundary condition – in a real device, there is a significant geometry dependence.
  - Neutron multiplication in structural materials has implications for tritium breeding.
    - Design requirements for a neutron multiplier (e.g., $TiBe_{12}$) may be eased by the right choice of SM.
- Impact of delayed radiation back-fluxes?
  - Over time, a fusion power reactor will accumulate significant concentrations of radionuclides.
  - We consider the build-up of back-fluxes from these in the following slides…

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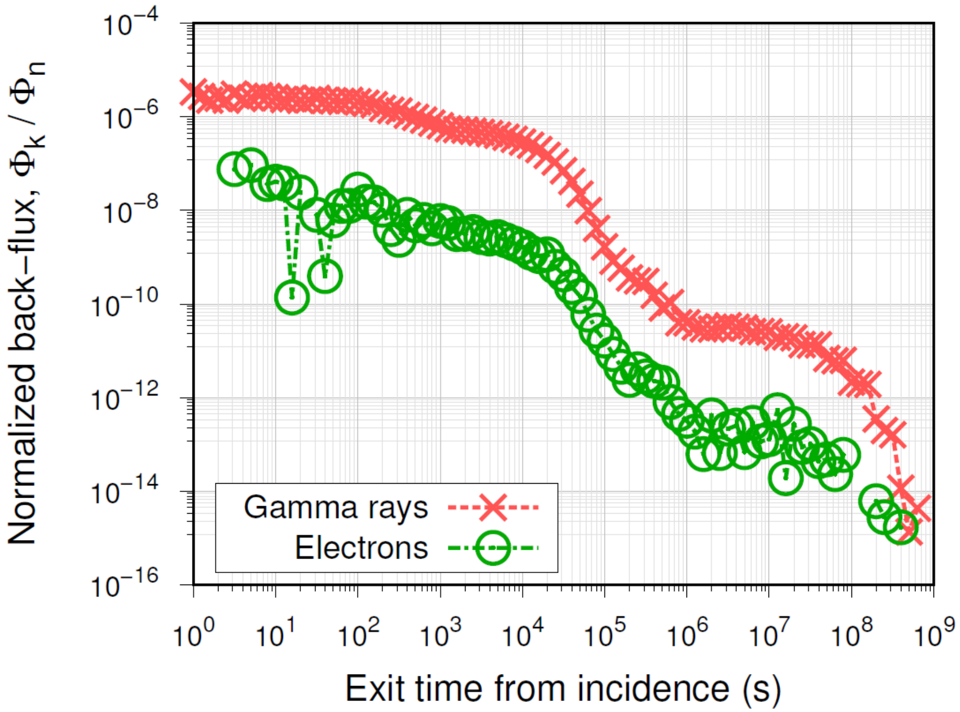
Analyzing the build-up of delayed radiation back-fluxes



Time-resolved delayed gamma ray and electron back-fluxes from a single neutron incident on RAFM steel with 2-mm FW thickness. These data can be obtained using the `ACT` *card.*

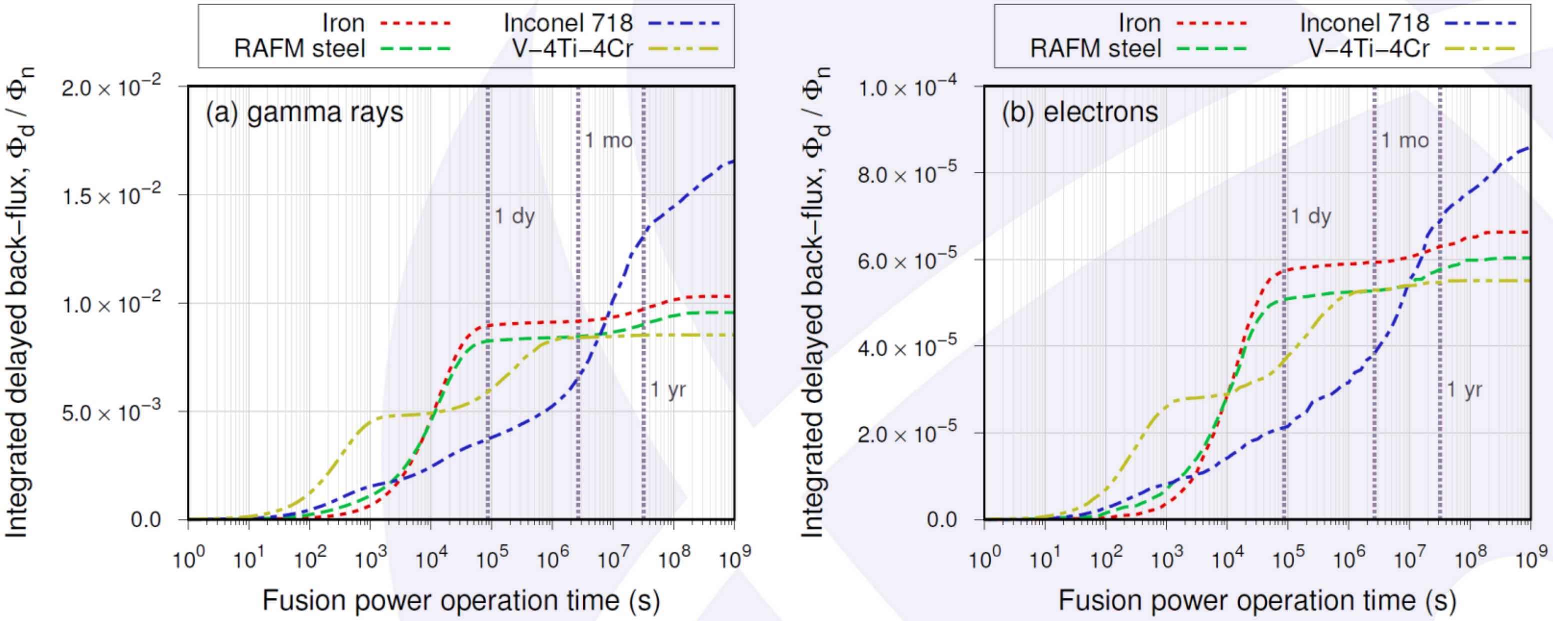
- MCNP output: **`F1`** tally with ~180 logarthmically-spaced time bins from $t = 10^{-9}$ s to $t = 10^{9}$ s.
  - Normalize tallies to fluxes by dividing by time bin width δt.
- Calculating the back-flux build up from steady-state neutron irradiation during fusion power operations:
  - Let $\nu(\Delta t - t')$ be the differential back-flux some time Δt after turning on the reactor from a fusion neutron incident at time t'.
    - Plot on left.
  - Total built-up delayed back-flux from all neutrons $0 \leq t < \Delta t$ is

$$\phi_\mathrm{d}(\Delta t) = \phi_\mathrm{n} \int_0^{\Delta t - t_\mathrm{c}} \nu(\Delta t - t')\, dt'$$

    - Prompt cutoff time $t_\mathrm{c} = 0.1$ ms.
  - Terminal back-flux level is approached as $\Delta t \to \infty$.
- We observe **no delayed neutron back-fluxes!** Only gamma radiation and electron emission.

Trends in built-up delayed radiation back-fluxes with materials design parameters

- SM selection is key.
  - Iron-based materials reach terminal back-flux levels within ~10 years of power operations.
  - V-4Ti-4Cr reaches terminal levels within about a year and looks like the best candidate.
  - Inconel 718 starts off well, but the build-up continues to the highest levels – no terminal back-flux level within ~30 years of operations.
    - Inconel 718 is probably the best structural material for a device you don't expect to work.
    - E.g., ITER.
- Effect of FW thickness (not shown) is the same as before – strong attenuation by high-Z.



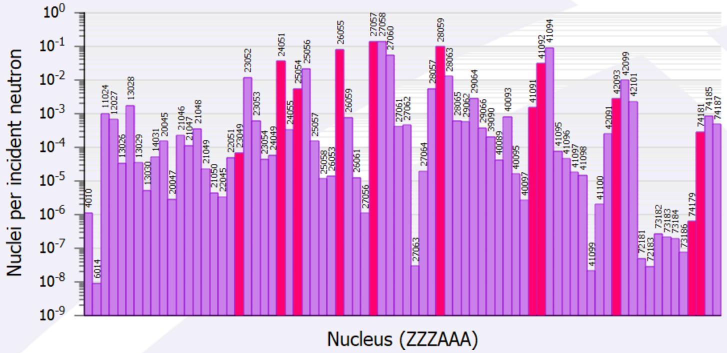
Time-integrated delayed back-fluxes of (a) gamma rays and (b) electrons for various structural materials with a 2-mm first wall thickness.

Implications of built-up delayed back-fluxes on fusion reactor operations

- Under normal steady-state operating conditions, delayed back-fluxes simply add to the total.
  - Terminal built-up levels reach 2-7% of the prompt back-fluxes, depending on material configuration.
- Delayed back-fluxes persist when fusion is “turned off” – during transients, most critically disruptions.
- Gamma radiation out of the walls can Compton scatter cold plasma electrons to high-energy runaways.
  - Compton cross section $\sigma_\mathrm{c} \leq \sigma_\mathrm{T} = 6.65 \times 10^{-29}\ \mathrm{m}^2$, electron density $n_\mathrm{e} \sim 4 \times 10^{20}\ \mathrm{m}^{-3} \ \rightarrow \lambda_\mathrm{mfp} \sim 10^7\ \mathrm{m}$.
  - ITER example: major radius $R_0 = 6.2\ \mathrm{m}$, Compton scatter probability $P_\mathrm{c} = 1 - \exp\left(-\frac{2R_0}{\lambda_\mathrm{mfp}}\right) \sim 10^{-7}$.
  - Fusion neutron production rate $R_\mathrm{f} \sim 10^{20}\ \mathrm{s}^{-1}$, terminal gamma radiation rate $R_\gamma \sim 10^{18}\ \mathrm{s}^{-1} \ \rightarrow R_\mathrm{c} \sim 10^{11}\ \mathrm{s}^{-1}$.
  - Sufficient runaway seeding to cause avalanche multiplication.
- Electron emission of highly relativistic ($E_\mathrm{se} > 100\ \mathrm{keV}$) electrons $R_\mathrm{se} \sim 10^{16}\ \mathrm{s}^{-1}$.
  - Even if most are trapped by the magnetic field, again enough can reach the plasma to induce the avalanche.
- Fully comprehending these effects requires scaling up to reactor geometries.
  - **`ACT`** card is too slow in MCNP6.3 for efficient large-scale simulations (improvements are coming, though!).
  - An explicit calculation approach based on the **`FT8 RES`** tally treatment may scale up more efficiently.

Progress towards an explicit calculation of the activated decay source distribution for delayed back-flux simulations

- At a high level this is a four-step process:
  1. Run neutron irradiation calculations in MCNP with tally **`F8`** and tally treatment **`FT8 RES`** to obtain the distribution of residual nuclides from nuclear activation.
  2. Use a decay solver to compute radioisotope concentrations $\{N_k(\Delta t)\}$. This corresponds to $\nu(\Delta t - t')$ from before.
     - Up to the present, our team used our own internal code for this.
     - Going forward, we would like to leverage CINDER 2024 instead, which will contain numerous data and algorithm improvements.
  3. Carry out the time-integration over operating time Δt. This corresponds to $\phi_\mathrm{d} = \int \phi_\mathrm{n} \nu(\Delta t - t')\, dt'$ from before.
  4. Convert the built-up radionuclide distribution into a decay source distribution (**`SDEF`** and associated cards) and run a follow-on MCNP calculation.
     - Since the emitted energy distributions depend on both particle type and geometry, the formulation of the source term is a bit complex. Examples on following slides.



An example of the radionuclide distribution in Inconel 718 with a 2-mm FW thickness obtained by the **`FT8 RES`** *treatment.*

- Initial investigations have succeeded in implementing this process, but the results are of limited validity due to nuclear data limitations (i.e., the **`cindergl.dat`** file).
  - Beta decay treatment is incomplete (lacking quantum data to convert discrete lines into continuous spectra).
  - Positron emission rates are not present – important for $\mathrm{e}^- + \mathrm{e}^+ \rightarrow 2\gamma$ annihilation photons.

Examples for decay source distribution in MCNP (1/2)

Transform 2D (c, p) index into 1D index!

```
c Source definition
sdef    ara=d1 par=fara=d2 erg=fara=d3 cel=fara=d6 x=0 y=0 z=fara=d9 wgt=1.403868
c ARA dummy distribution
si1     L   16  12  26  22  36  32  ...
sp1     D   0.00107  0.00133  0.133  0.00621  0.0325  0.00154  ...
c Particle type distribution
ds2     L   p   e   p   e   p   e   ...
c Cell distribution
ds6     L   100  100  200  200  201  201  ...
c Position distributions
ds9     S   19  19  29  29  39  39  ...
si19    H   0.0  0.2
sp19    D   0.0  1.0
si29    H  -1.0  0.0
sp29    D   0.0  1.0
si39    H  -1.25893  1.0
sp39    D   0.0  1.0
...
c Energy distributions - per cell and per particle
ds3     S   16  12  26  22  36  32  ...
si16    H   0.01  0.0125893  0.0158489  0.0199526  0.0251189  0.0316228  ...
sp16    D   0  0  0  4.34928e-008  0  2.45033e-007  ...
si12    H   0.01  0.0125893  0.0158489  0.0199526  0.0251189  0.0316228  ...
sp12    D   0  0  0  0  0  0  ...
...
```

ARA is only used for certain detector tallies, otherwise it has no effects!

Examples for decay source distribution in MCNP (2/2)

```
c Source definition
sdef    cel=d1 par=fcel=d2 erg=fcel=d3 x=0 y=0 z=fcel=d9 wgt=1.403868
c Cell distribution - simpler than using a dummy variable but maybe a bit more obtuse
si1     L   100  100  200  200  201  201  ...
sp1     D   0.00107  0.00133  0.133  0.00621  0.0325  0.00154  ...
ds6     L   100  100  200  200  201  201  ...
c Particle type distribution
ds2     L   p   e   p   e   p   e   ...
c Position distributions
ds9     S   19  19  29  29  39  39  ...
si19    H   0.0  0.2
sp19    D   0.0  1.0
si29    H  -1.0  0.0
sp29    D   0.0  1.0
si39    H  -1.25893  1.0
sp39    D   0.0  1.0
...
c Energy distributions - per cell and per particle
ds3     S   16  12  26  22  36  32  ...
si16    H   0.01  0.0125893  0.0158489  0.0199526  0.0251189  0.0316228  ...
sp16    D   0  0  0  4.34928e-008  0  2.45033e-007  ...
si12    H   0.01  0.0125893  0.0158489  0.0199526  0.0251189  0.0316228  ...
sp12    D   0  0  0  0  0  0  ...
...
```

Conclusions and Future Work

- Used MCNP simulations to quantify the radiation back-fluxes from neutron irradiation in fusion reactors.
  - Neutron back-fluxes ~60-90% of incident flux – this is an unexpectedly large value!
  - Gamma radiation back-fluxes ~15-50% of incident flux.
  - Electron back-fluxes ~2 orders of magnitude lower than gamma ray back-fluxes.
- Characterized the impact of materials design configuration on back-fluxes.
  - Neutron back-flux magnitude depends primarily on the choice of structural material.
  - Gamma ray and electron back-fluxes depend primarily on tungsten first wall thickness.
- Computed the time-integrated build-up of delayed back-fluxes from radioactive decay.
  - Delayed gamma ray and electron back-fluxes ~2-7% of the prompt back-flux magnitude.
  - Iron-based materials and V-4Ti-4Cr reach terminal build-up levels; Inconel 718 does not (within ~30 years).
- Large back-flux magnitudes have significant implications for power handling and disruption mitigation.
- Future work: scaling up to perform reactor-relevant calculations!
  - Real reactor geometries require HPC resources
  - Explicit delayed back-flux solution with decay source distribution for better efficiency.
  - Coupling to plasma physics to quantify effects on plasma performance and disruption handling.