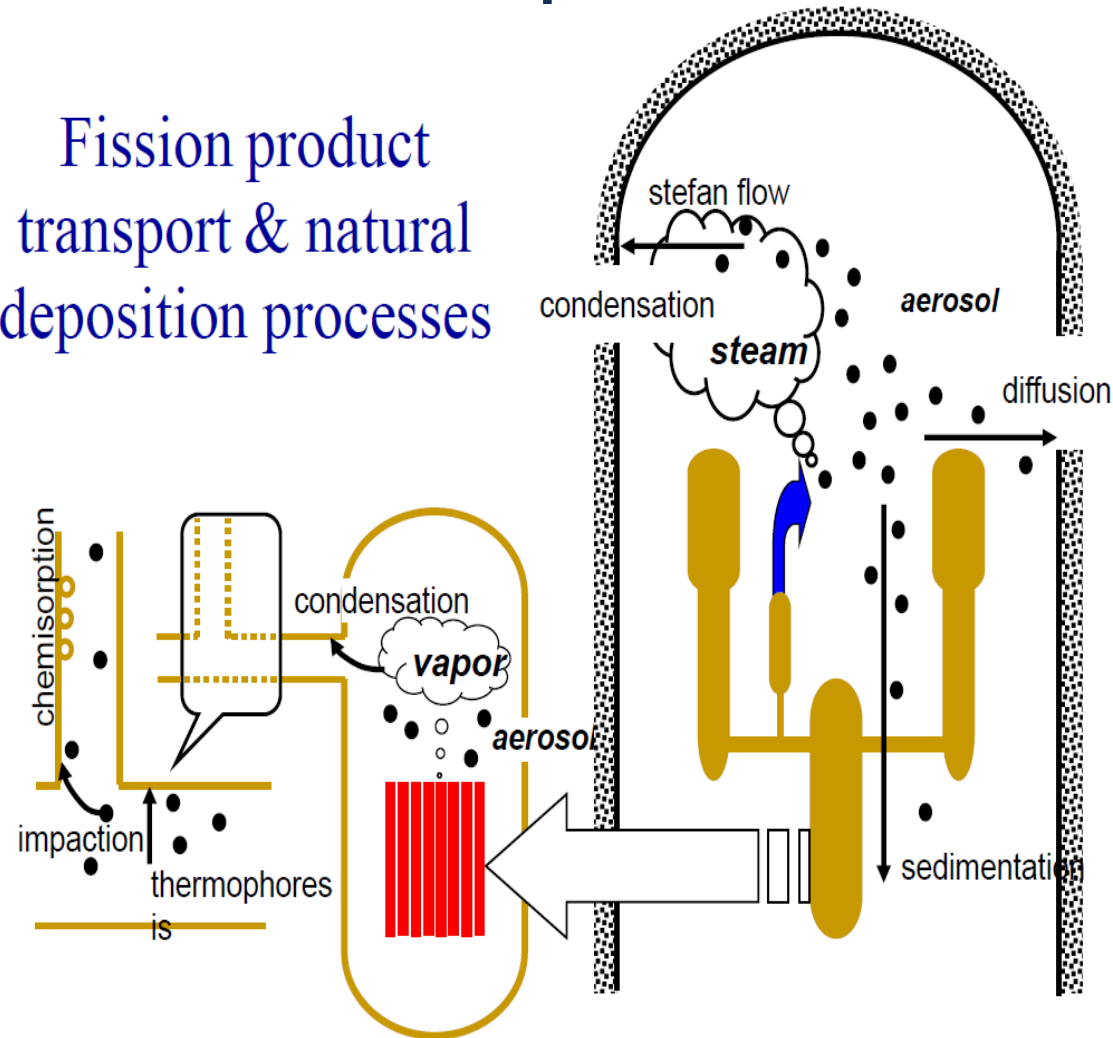




# Why use MELCOR for dose approximations in a nuclear plants

## Fission product transport & natural deposition processes

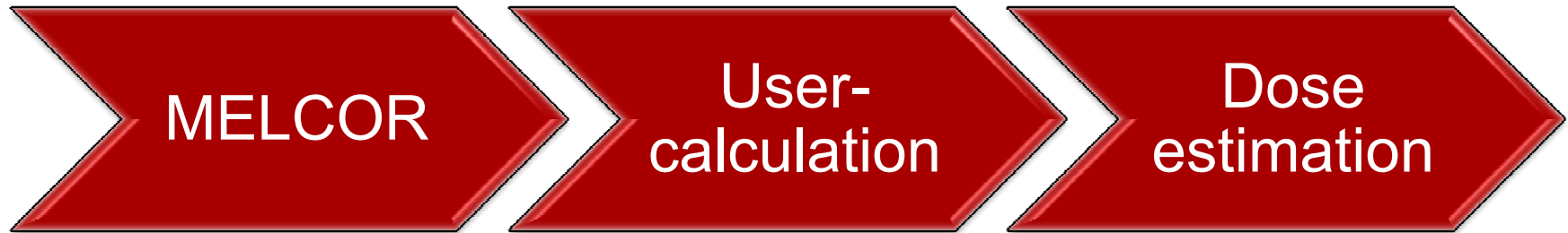


- Take advantage of *integrated* physics in MELCOR
- Detailed prediction of the time-history of:
  - Integrated plant thermal-hydraulics (RCS, containment, buildings)
  - Integrated transport of fission products
- Use MELCOR alone to approximate dose calculations, couple to other codes for more detail

# Using MELCOR to calculate dose

- MELCOR provides a mechanistic calculation of the thermal-chemical transport of *radionuclides* throughout a nuclear power plant: core, vessel, RCS, containment, auxiliary buildings.
- MELCOR radionuclide classes are elements lumped chemically.
- MELCOR does not normally track individual isotopes.
- However, given user-knowledge of the detailed elemental and isotopic breakdown of each MELCOR RN class, MELCOR can accurately estimate the activity distribution of radioisotopes of interest to the dose calculation.
- **Most significant approximation**: estimating dose (or exposure) given MELCOR-calculated radionuclide activities/masses
  - Dose/exposure depends on radiation *particle* (gamma/beta) flux and energy spectrum – *MELCOR does not model particle transport*

# User responsibility in MELCOR dose calculation



Internal MELCOR physics simulates the transport of radionuclides throughout the nuclear plant. Given user-knowledge of detailed inventory and fission product decay chains, MELCOR can accurately estimate activities of particular isotopes.

User-specified calculation interface from radioisotope activity to dose:

- Closed-form analytical expressions for particle transport to a detector using CF package
- External coupling to another code

Given flux distribution and energy spectrum of gamma and/or beta radiation field, MELCOR can estimate dose, dose rate, and exposure (ionization) rate in any region of the plant:

- Containment
- Auxiliary buildings

# MELCOR dose calculation process

- **Select isotopes/isomers of interest to dose calculation**
  - For example, if only concerned about gamma emitters, neglect alpha and beta emitters.
- **For each MELCOR chemical group released, break down to individual isotope and convert mass to units of activity**
- **Adjust activity to account for radioactive decay over the period of dose assessment**
- **Develop a set of assumptions that could simplify calculation within reasonable limits (e.g., source geometry, spatial distribution of activity, location, shielding)**
- **Use derived equations to determine gamma or beta flux and dose rate based on assumed geometry for each gamma or beta emitters considered**
- **Determine the dose rate or the total integrated dose**

# MELCOR calculation of activity: simple user-intensive option

- **Converting RN class mass released to a MELCOR control volume into activity: first calculate isotopic masses**

$$M_i = M_{RN,x} \frac{M_i}{M_x}$$

Here,  $M_i$  is the mass of isotope  $i$ ,

$M_{RN,x}$  is the mass of MELCOR RN class  $x$  released to the region of interest,

$M_i$  is the initial core inventory at shutdown of isotope  $i$  which is obtained from ORIGEN, and

$M_x$  is the initial core inventory at shutdown of MELCOR RN class  $x$ .

- **Next, convert isotopic masses to activity using specific activity for each isotope**

$$\text{Specific activity} = \lambda N = \frac{0.693}{\dots} * \frac{6.022 \times 10^{23}}{\dots} = \frac{4.18 \times 10^{23}}{\dots}$$

# Limitations of simple user-intensive option

- **MELCOR RN does not account for radioactive decay – activity is associated with fission products themselves and does not include activity produced by decay of daughters (RG.189 App. D applies a correction factor of 1.3 to account for decay chains)**
- **Simple user-estimation of isotopic activities has limited accuracy**
  - No accounting of radioactive decay chains or thermal neutron capture
- **Requires lots of control functions to be added to models**
- **Use BONUS code added to MELCOR 2.1 in 2010 for more formal and high accuracy calculation of isotopic activities**
  - provides calculations of FP evolution after reactor shutdown
- **Both methods required detailed inventory data calculated by neutronics codes (e.g. ORIGEN)**

# MELCOR calculation of activity: BONUS model in MELCOR

- **Allows MELCOR to take into account beta and gamma decay chains of fission products after reactor shut down**
  - no change in mass number and FP decay sequence represents the isobaric chain.
  - $\beta$ -decay assumes one or two daughter nuclides: ground and isomeric states
- **Thermal neutron capture is taken into account**
  - produces one or two daughter nuclides with a mass number increased by 1
  - results in coupling of the current chain with the next chain(s).
- **Requires initial isotopic composition of burned fuel as always**
- **BONUS model also requires some additional MELCOR input:**
  1. Fission product properties: half-life times, decay energies
  2. Input for decay chains
  3. Additional input controlling activity calculation: specify which control volumes and/or heat structures to turn on BONUS model

# Calculate dose given MELCOR-supplied activities throughout plant

## *One possible method*

- **Gamma dose rate in air**

$$\dot{D}_i^\gamma = (5.77 \times 10^{-5}) \phi_i E_i \left( \frac{\mu_{a,i}}{\rho} \right)_{air}$$

- $\dot{D}_i^\gamma$  - gamma dose rate of the  $i^{\text{th}}$  isotope [Rad/hr]

- $\phi_i$  - gamma flux of the  $i^{\text{th}}$  isotope [ $1/(\text{cm}^2 \text{ s})$ ]

- $E_i$  - gamma energy of the  $i^{\text{th}}$  isotope [MeV]

- $\left( \frac{\mu_a}{\rho} \right)_{air}$  - mass absorption coefficient for air at  $E_i$  [ $\text{cm}^2/\text{g}$ ]

- **Assume mono-energetic gammas for each isotope**

- In reality, gammas interactive with matter before reaching detector, creating a energy spectrum

- **For each isotope of interest, this approach amounts to estimating the gamma flux at a detector**

# Flux estimations for dose calculation Sandia National Laboratories

- **In general, a 3-D particle transport calculation is necessary for high accuracy calculation of energetic particle flux at a radiation detector**
  - Depends on the detector design, 3-D arrangement of obstructions in the plant region interest (metal and concrete between source and detector)
    - ❖ **Scattering, absorption, and secondary particles can affect flux by orders of magnitude**
  - Time-dependent thermal-hydraulics and radionuclide transport
  - Requires external coupling to another code
- **For first order approximations using only MELCOR, implement simple user-specified model for particle flux**
  - Assume a geometric distribution of activity in the MELCOR control volume(s) of interest
  - May consider airborne activity, activity in liquid pools, activity deposited on the floor, and activity plated out on vertical heat structures
  - User choice to neglect/include various contributions to total flux

# Un-collided gamma flux, spherical distribution of gamma-emitting radionuclides

- **For a uniform and spherical distribution of activity with a detector located at the center of the sphere:**

$$\phi_i = (3.7 \times 10^{10}) \frac{A_i}{V \rho_{air} \left( \frac{\mu_s}{\rho} \right)_{air}} \left( 1 - e^{-\rho_{air} R \left( \frac{\mu_s}{\rho} \right)_{air}} \right)$$

- $A$  = activity for the  $i^{\text{th}}$  isotope (Ci),
- $R$  = equivalent spherical radius of volume  $V$  ( $\text{cm}^3$ ),
- $\rho_{air}$  = density of air ( $\text{g}/\text{cm}^3$ ),
- $\left( \frac{\mu_s}{\rho_{air}} \right)$  = total attenuation coefficient of air at  $E_i$  ( $\text{cm}^2/\text{g}$ ),
- and  $V$  = volume of MELCOR control volume(s) of interest ( $\text{cm}^3$ ).
- **Assumptions:**
  - Mono-energetic flux
  - Mono-energetic gamma particles from each isotope reach the detector
  - attenuation coefficient: consider energy dependence or treat as a constant

# Attenuation coefficient – function of energy

- $$\left(\frac{\mu_a}{\rho}\right)_{air} = a + bE + c[\ln(E)]^2 + d\sqrt{E} + e[\ln(E)]$$
$$\left(\frac{\mu_s}{\rho}\right)_{air} = a + bE^c$$

fit coefficient	mass absorption coefficient for air	mass attenuation coefficient for air
a	0.0	-0.037274
b	-5.2588E-04	0.101714
c	-5.2077E-03	-0.274123
d	2.8172E-02	--
e	-1.7809E-02	--

Simplest approach: energy is the initial energy of the emitted gamma for each isotope (effectively giving each isotope its own distinct attenuation coefficients).

# Un-collided gamma flux due to activity on control volume floor

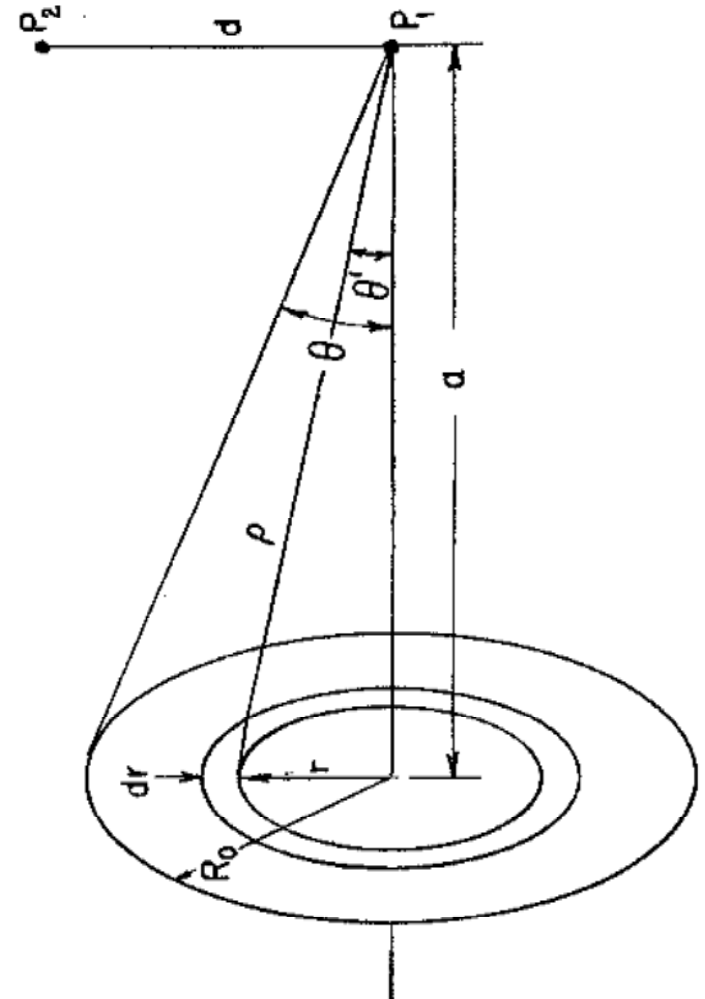
- Simplest case: detector at centerline

$$\phi_i = \frac{A_i}{2 \cdot \pi \cdot R_o^2} [E_1(b_1) - E_1(b_1 \sec \theta)]$$

$$E_n(b) = b^{n-1} \int_{b_n}^{\infty} \frac{e^{-t}}{t^n} dt \text{ (Exponential integral)}$$

- **Numerous other analytical and computational models available to estimate un-collided flux, some more complicated than others...**

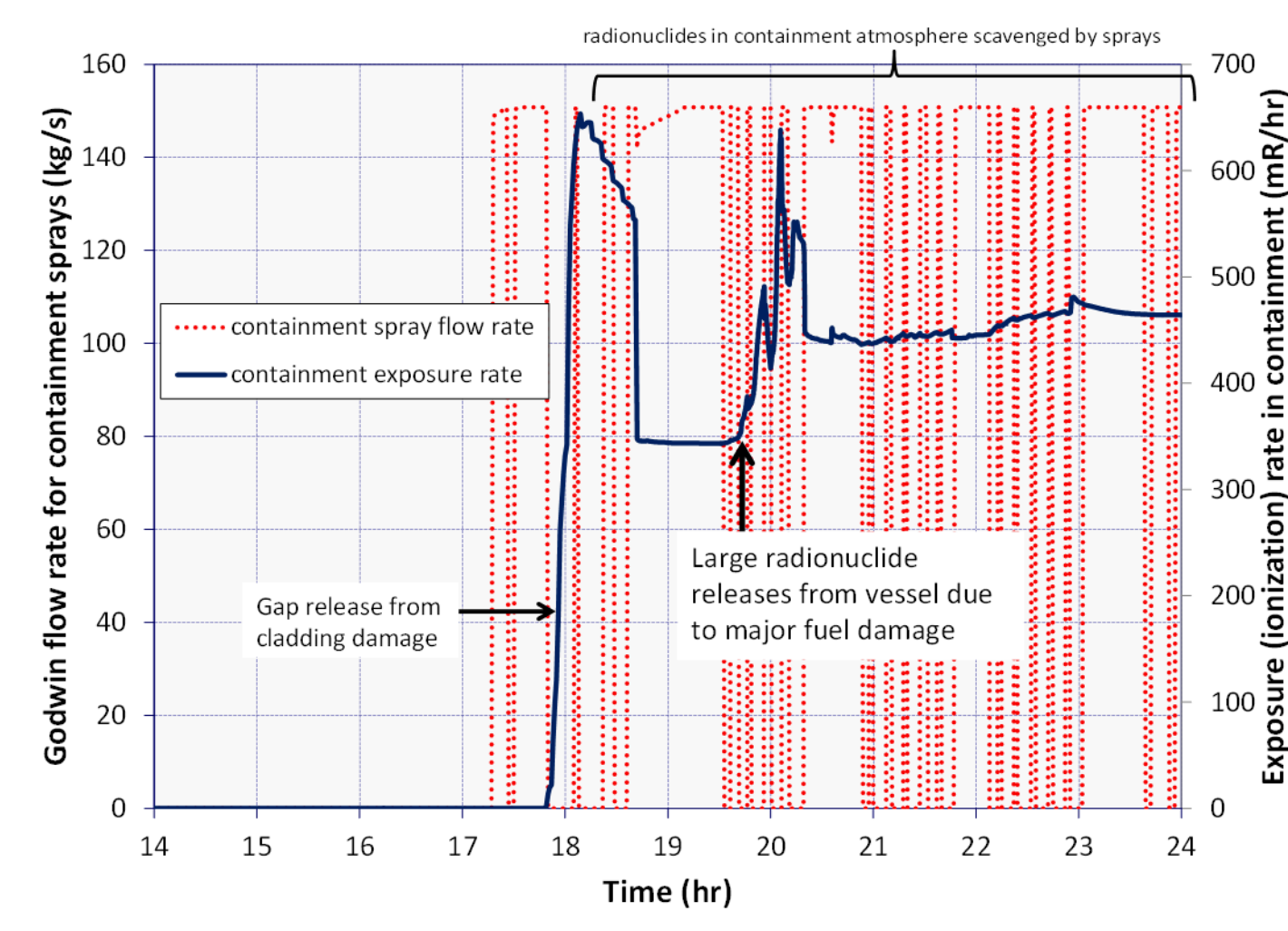
- Sources in pools
- Sources on horizontal slabs
- Cylindrical sources in pipes
- Detectors in complicated orientations with source



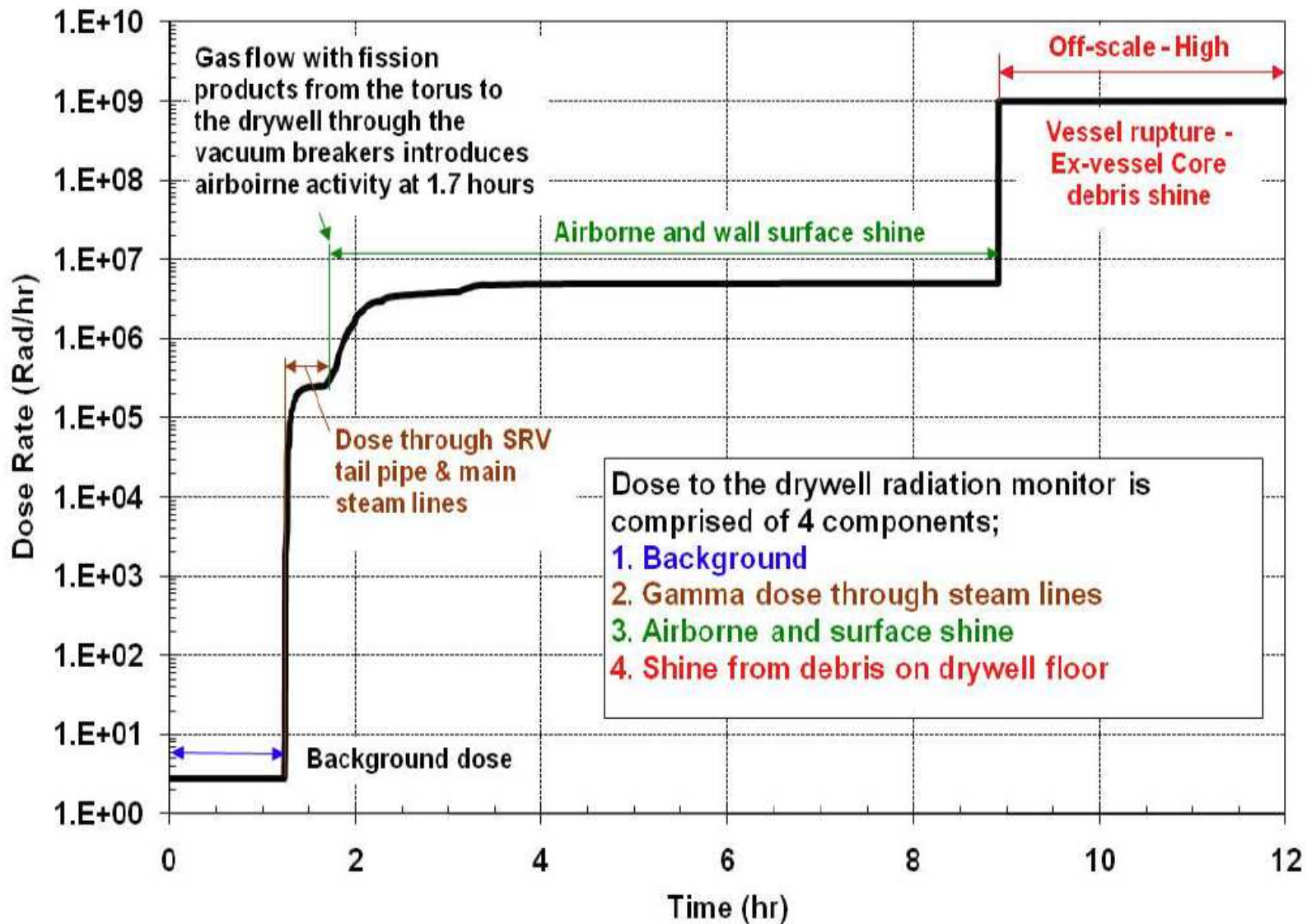
# How to implement dose calculations in MELCOR

- **External code coupling**
  - most complicated, most CPU intensive
- **Implement flux and dose calculations via MELCOR control functions and tabular functions**
  - Make use of FORMULA control functions, external data files, etc.
  - May be cumbersome for dose calculations in several regions and for lots of isotopes
- **Implement flux and dose calculations using user-defined functions**
  - Fortran subroutines added to MELCOR code upon compile time
  - The functions are called from special MELCOR control functions
  - Requires a special executable to be used with the MELCOR input deck that uses the user-defined function
- **Post-process MELCOR data to determine fluxes and doses**

# MELCOR estimation of exposure rate in containment atmosphere (PWR)



# BWR SBO: drywell dose rate



# MELCOR dose work at Sandia

- **Preliminary investigations by Raymond Jun**

R. Jun, “Application of MELCOR to Containment Dose Analysis,” *Proceedings of CSARP 2010*, Bethesda, MD, September 16-17, 2010.

- **Proposed Equipment Dose Calculation Method for the Regulatory Guide 1.89 Update by Donald Kalinich**

D. Kalinich, et al., “Proposed Equipment Dose Calculation Method for the Regulatory Guide 1.89 Update,” DRAFT, 2012.

- **Regulatory Guide 1.89, Revision 1, “Environmental Qualification of Certain Electrical Equipment Important to Safety for Nuclear Power Plants”**

# Purpose

- Recommend revisions to containment equipment dose rate calculation methods based on review of the research insights gained since RG 1.89 was issued.
- Develop a simple/hand-calculation method for determining containment equipment dose rates.

# Summary of Calculation Method Revision Recommendations

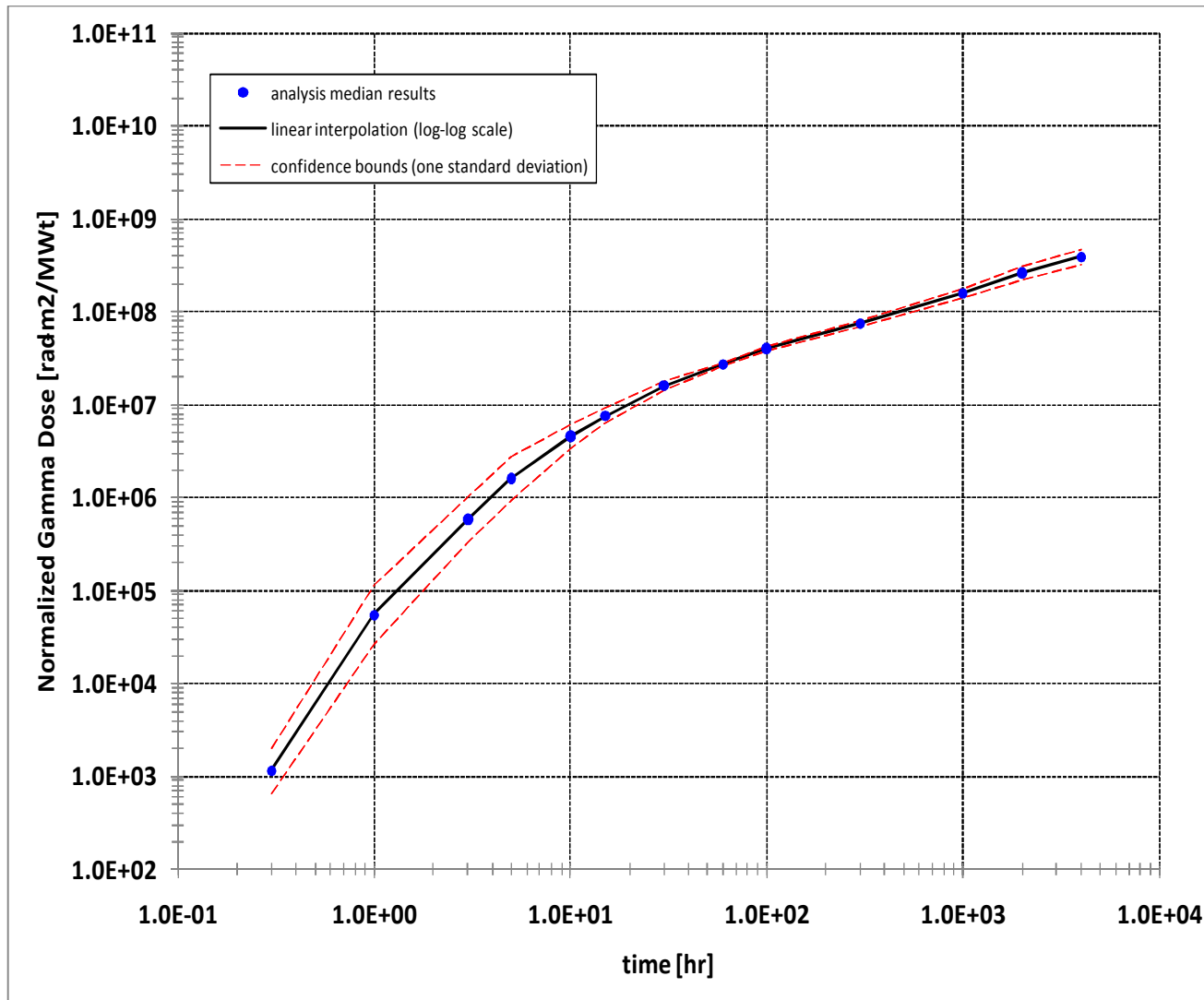
attribute	RG 1.89	calculation recommendation
containment source term	TID-14844	MELCOR-calculated source term
in-containment source term transport	TACT	MELCOR-calculated in-containment source term transport
In-containment source term removal mechanisms	Series of side calculations (e.g. SPIRIT, hand calculations) for removal rates	MELCOR-calculated containment source term removal
beta integrated dose rates	airborne: well-mixed infinite medium surfaces: none pool: none	use the “same” methods as used in RG 1.89 and RG 1.183
gamma integrated dose rates	airborne: well-mixed cylinder volume surfaces: cylinder(?) surface pool: large (infinite?) pool	use the “same” methods as used in RG 1.89 and RG 1.183

“same” is designated as no guidance provides the specific dose rate equations or supporting parameters (e.g., mass attenuation and absorption coefficients)

# Proposed Simple/Hand-Calculation Integrated Dose Rate Calculation Method

- 1) for each accident sequence important to risk, use MELCOR to determine the source term release to the containment
- 2) for each accident sequence important to risk, use MELCOR to calculate transport of the source term through the containment, accounting for removal mechanisms
- 3) for each accident sequence important to risk, use the MELCOR result from (2) to calculate beta and gamma dose integrated dose rates
- 4) apply the statistical method used in the SNL NUREG-1465 update analysis to create representative beta and gamma integrated dose rate time-histories.
- 5) scale the representative beta and gamma integrated dose rate time-histories by the appropriate reactor parameter(s) (e.g., containment volume, reactor power)

# Integrated PWR containment dose



# Summary

- A simple general approach can be used to approximate dose using MELCOR source term, with limitations: Results will be sensitive to:
  - Nodalization scheme, depending on the region of interest (e.g., a single volume vs. multi-compartment nodalization)
  - Geometry assumption in particle flux and transport calculations
  - Number of isotopes considered, gamma energy assumptions
- MELCOR can be used to calculate accurate isotopic activities throughout the plant, given sufficient user input
- Nonetheless, use of MELCOR source term in dose assessments such as equipment qualification could provide means to reduce conservatism with a deterministic model for FP transport and deposition in the containment

# EXTRA SLIDES

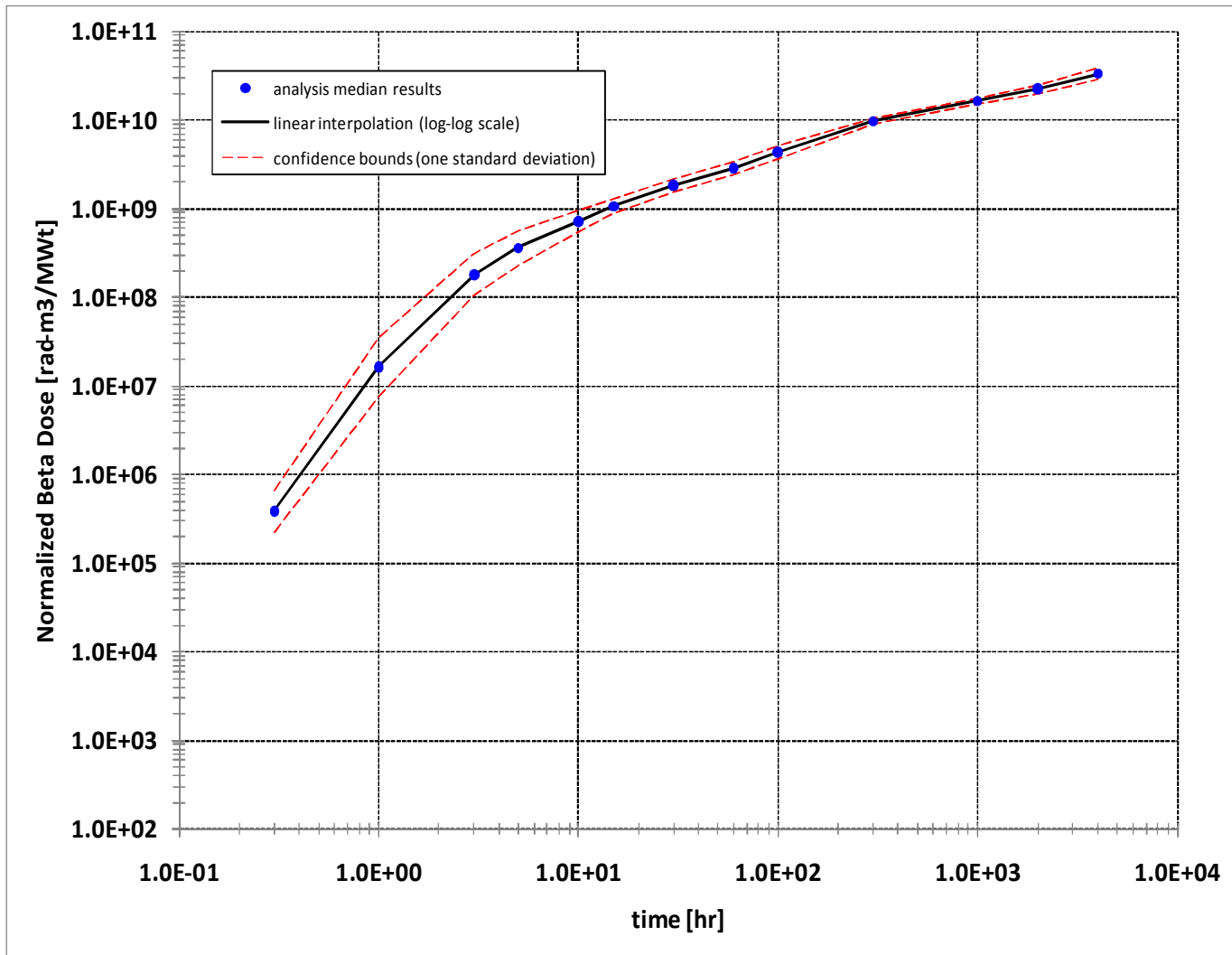
# Beta dose calculations

- The beta dose rate in air from an infinite volume with a uniform activity concentration is given as

- $$\dot{D}_i^\beta = (2.13 \times 10^6) \frac{A_i \bar{E}_{\beta_i}}{V \rho_{air}} \quad (13)$$

- $\dot{D}_i^\beta$  - beta dose rate of the  $i^{\text{th}}$  isotope [Rad/hr]
- $\bar{E}_{\beta_i}$  - average beta energy of the  $i^{\text{th}}$  isotope [MeV]
- In the beta dose rate calculation, radioactive mass contained in pools was ignored in the determination of the activity concentration.

# PWR SBO containment beta dose



# Dose calculations

- The response of radiation detectors during a severe nuclear accident is a complex problem that depends on several variables, such as the specifics of the detector design, the 3-D geometry of the plant, shielding (such as metal and concrete obstructions) between the radiation source and the detector, the reactor thermal-hydraulics, and the transport of radionuclides throughout the plant. A mechanistic analysis of detector behavior would require a coupled radionuclide and radiation transport simulation, e.g. MELCOR for radionuclide transport inside containment and the auxiliary building and MCNP for gamma particle transport to the detector. Nevertheless, simple analytical formulas for gamma transport can be used in conjunction with MELCOR in order to provide first-order approximations of radiation detector behavior during severe accidents.
- Dose or exposure calculations are complicated since these quantities depend on the gamma flux, gamma energy, and the multidimensional transport/attenuation/backscatter of gamma particles through plant structures and atmospheres. Several simplifying assumptions must be made in order to calculate the dose or exposure rate using only MELCOR control functions for a given plant-distribution of chemical radionuclide masses calculated by MELCOR.