Hot Cell Window Shielding Analysis Using MCNP

Safety Analysis Working Group

Chad Pope
Wade W. Scates
J. Todd Taylor

May 2009

Idaho National Laboratory

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Chad Pope
Wade W. Scates
J. Todd Taylor
Idaho National Laboratory
P.O. Box 1625
Idaho Falls, ID 83415-6108
208-533-7745 office
208-533-7239 fax
chad.pope@inl.gov

INTRODUCTION

The Idaho National Laboratory Materials and Fuels Complex nuclear facilities are undergoing a documented safety analysis upgrade. In conjunction with the upgrade effort, shielding analysis of the Fuel Conditioning Facility (FCF) hot cell windows has been conducted. This paper describes the shielding analysis methodology and results.

FACILITY DESCRIPTION

A photo of FCF is shown in Figure 1. FCF is in the foreground with the Experimental Breeder Reactor II (EBR-II) in the background on the left. The two facilities are connected; this allowed spent fuel transfers directly from EBR-II to FCF prior to termination of EBR-II reactor operations in 1994.

Figure 1. Fuel Conditioning Facility.

FCF uses an engineering scale pyrometallurgical process to treat fast reactor spent fuel from EBR-II. Spent fuel from EBR-II is unique in that it is metallic rather than oxide and contains elemental sodium metal as a bonding agent between the fuel and stainless steel cladding.

The entire pyrometallurgical process is conducted remotely in a shielded hot cell environment. Figure 2 shows the hot cell layout. The rectangular portion of the hot cell contains an air atmosphere and is used for storing spent fuel. The annular portion of the hot cell contains an inert argon gas atmosphere and is used for all processing steps. An inert atmosphere is necessary because some steps in the process involve pyrophoric metals.
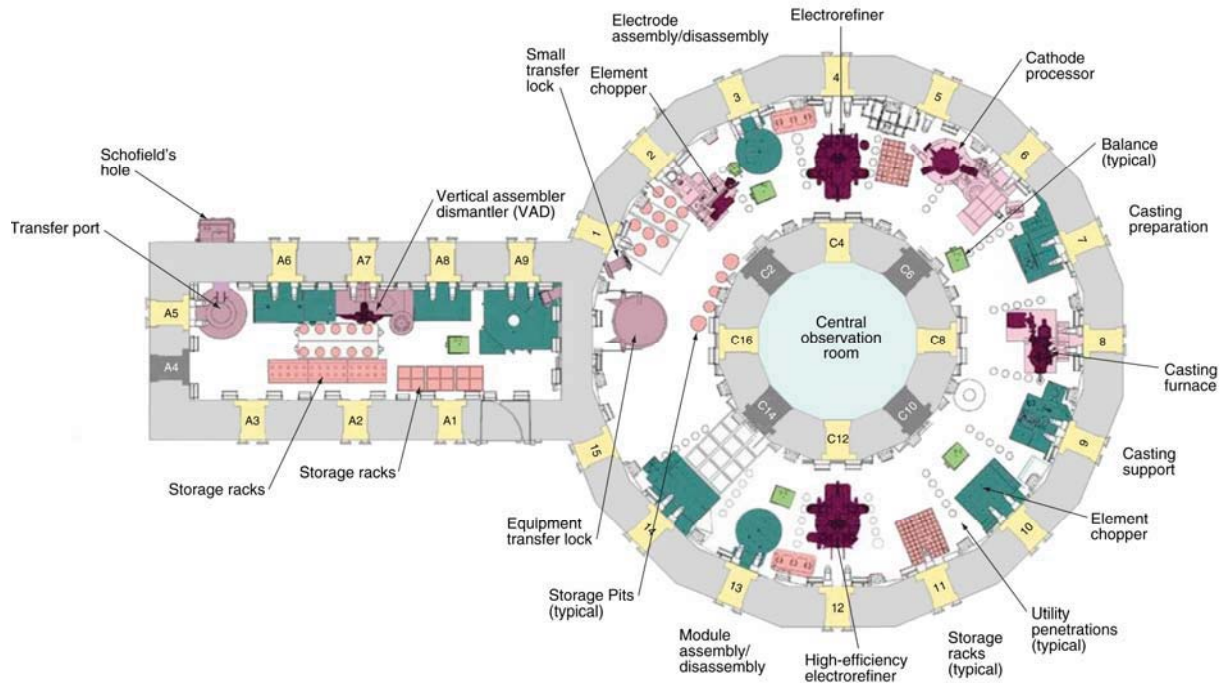

Figure 2. Facility Layout.

Shielding is primarily provided by 5-ft-thick, high density concrete walls. Shielding windows are used to allow equipment viewing and operation. A cross-section view of a shielding window is shown in Figure 3. The personnel access area resides on the left side of the figure while the radioactive material area resides on the right side of the figure. The window design is intricate and uses nine glass slabs and mineral oil between the slabs for optical clarity. Table 1 provides a listing of glass slabs that comprise the window.
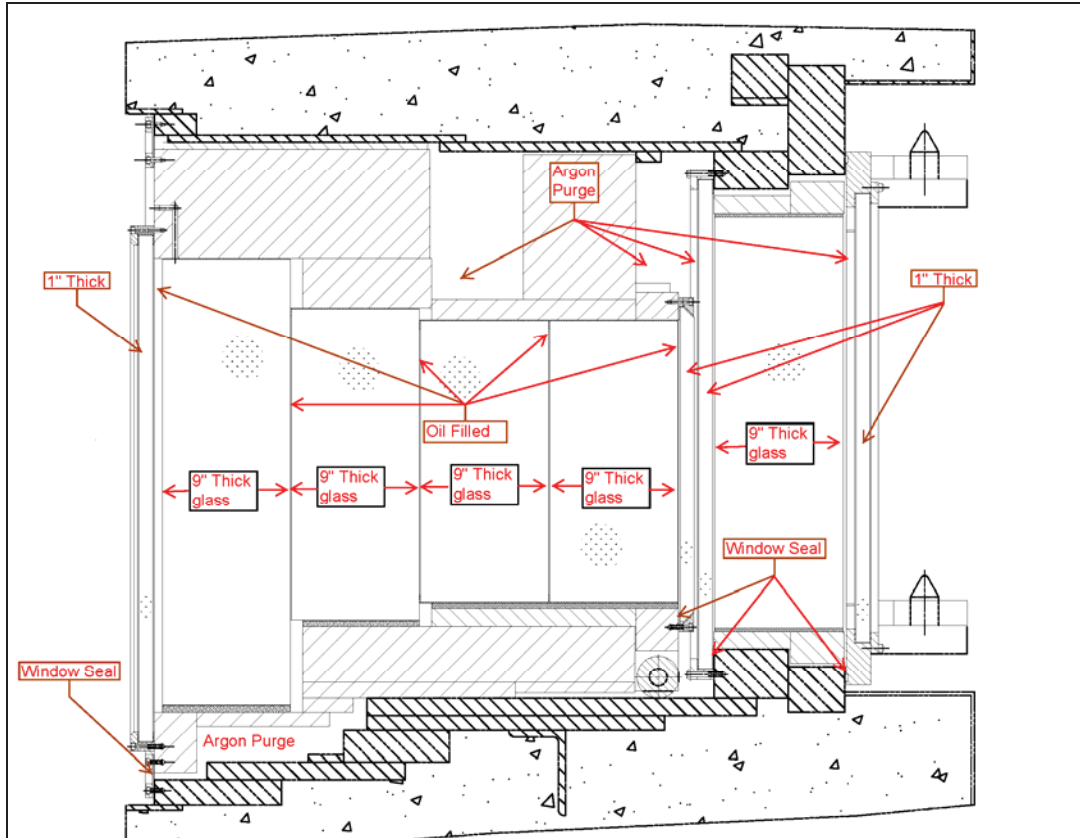
Figure 3. Window Side View.

Table 1. Window Slab Data.

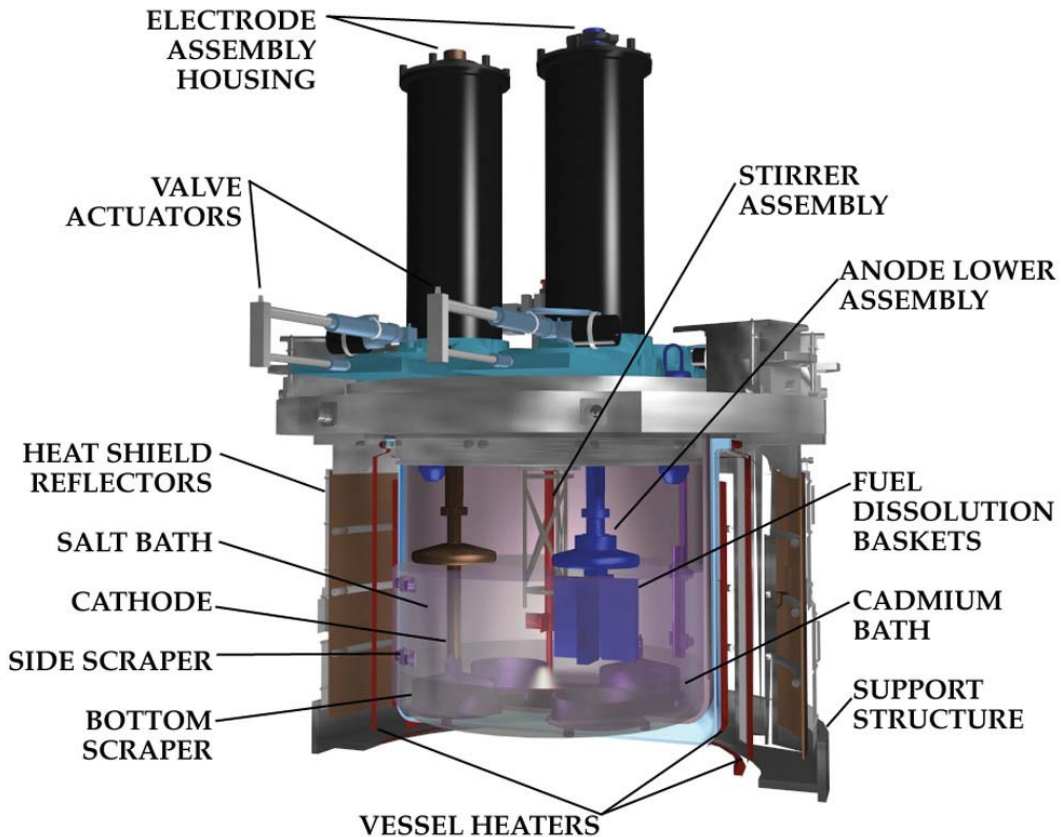| Slab Identifier (right-to-left)	Thickness (inches)	Density (g/cm^3)
A	1	2.7
B	9-1/16	3.3
C	1	2.7
D	1	2.7
E	9-1/16	3.3
F	9-1/16	3.3
G	9-1/16	3.3
H	9-1/16	3.3
J	1	2.5

Figure 4. Electrorefiner.

The spent fuel process concentrates fission products in a process vessel known as the electrorefiner (see Figure 4). In the electrorefiner, chopped spent fuel is immersed in molten LiCl-KCl salt. The active metal fission products, transuranic metals, and sodium metal in the spent fuel undergo chemical oxidation and form chlorides. Voltage is applied between the basket, which serves as an anode, and a mandrel, which serves as a cathode, causing metallic uranium in the spent fuel to undergo electro-chemical oxidation thereby forming uranium chloride. Simultaneously at the cathode, uranium chloride undergoes electro-chemical reduction and deposits uranium metal onto the mandrel. The uranium metal is subsequently removed from the electrorefiner and stored. Fission products primarily remain as chlorides in the salt. Over time, buildup of fission products in the electrorefiner salt produces the bounding gamma radiation source in the facility. The primary fission product of concern is ^{137}Cs, and the quantity of ^{137}Cs can potentially exceed several hundred thousand curries.

The other radiation source that must be considered in FCF is associated with an accidental criticality. Since the FCF process is conducted in an inert argon atmosphere, a bounding, unmoderated criticality was assumed to involve 10^{17} fissions.

MCNP Model

In order to calculate the potential dose from an accidental criticality or direct gamma radiation from the electrorefiner, a detailed MCNP (Version 5.1.40) model of the cell wall and window along with the radiation sources was developed. The calculations were performed using a Sun Workstation Model W2100Z with two AMD Optiron 200 series CPUs. The MCNP model geometry is shown in Figure 5. The gray rectangle represents the gamma radiation source from fission-product-laden molten salt in the electrorefiner. The electrorefiner is centered roughly 8 ft from the cell wall. Yellow in the figure represents argon gas. Red represents high density concrete. Blue represents 3.3 g/cm^3 density glass, and green represents 2.7 g/cm^3 density glass. Dark blue represents mineral oil. Orange represents steel slabs that make up the window tank unit. Pink represents the operator work area.
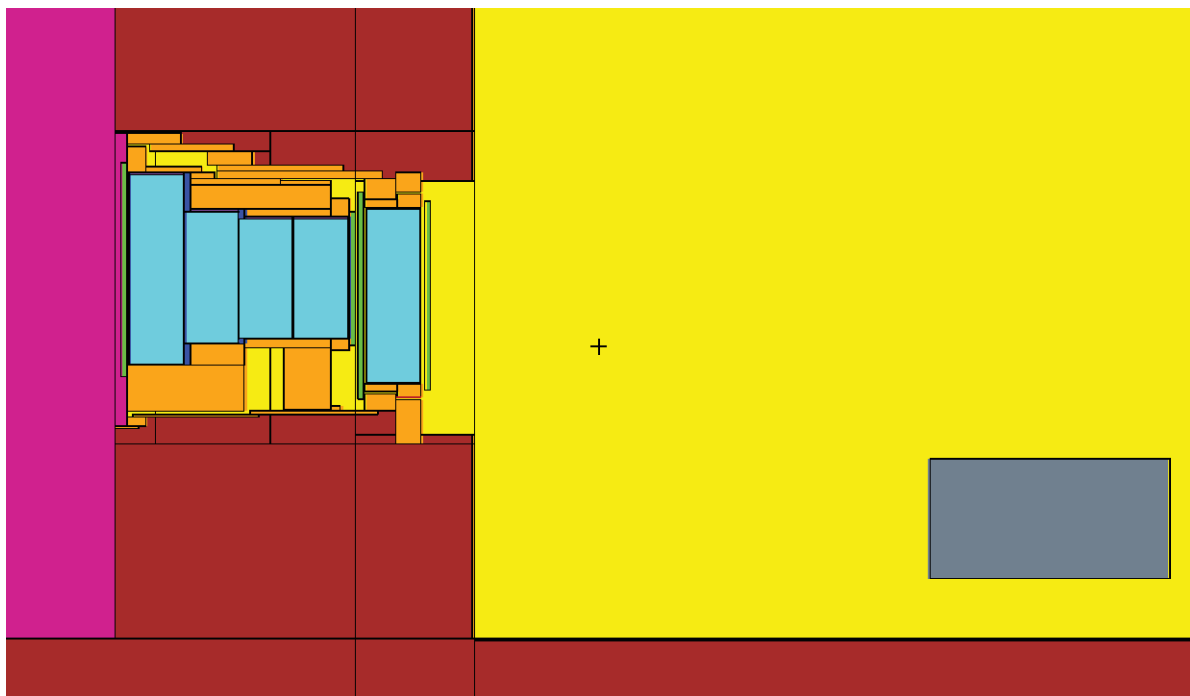

Figure 5. MCNP Model.

The dose calculations associated with an accidental criticality were performed using the MCNP KCODE option. The required information to determine the dose-in-air per fission includes:

ν, the average number of neutrons released per fission,

D_n, the dose-in-air due to neutrons at the detector location, and

D_γ, the dose-in-air due to photons at the detector location.

Both D_n and D_γ are determined using point detector (F5) tally cards. D_γ includes a prompt gamma contribution from the excursion and a secondary gamma contribution from gammas

generated during (n,γ) reactions. Neutron and photon flux-to-dose rate conversion factors (for the MCNP DE and DF cards) were obtained from Appendix H of the MCNP manual.[1]

The dose-in-air per fission is found by taking the average neutrons per fission and multiplying it by the dose-in-air per neutron:

$$D_f = v \cdot \left(D_n + D_\gamma \right)$$

The dose due to an accidental criticality can be found by multiplying the dose-in-air per fission by the number of fissions, FY_{max}:

$$D_{max} = D_f \cdot FY_{max}$$

This technique includes the dose due to prompt neutrons and gammas associated with the excursion and the secondary gammas generated by (n,γ). This methodology does not address the gammas associated with criticality generated fission products, which are negligible through the window.

The dose calculations associated with fission product gamma radiation source in the electrorefiner were performed using the MCNP SDEF option. A cylindrical volume source was distributed within the electrorefiner salt volume. The source particles were generated with uniform probability throughout the volume. Consistent with ^{137}Cs, the source particles were given an initial energy of 661.65 keV. Flux at the detector location was determined using a point detector tally card. As stated above, the photon flux-to-dose rate conversion factors were obtained from Appendix H of the MCNP manual.

RESULTS

To determine the maximum dose from an accidental criticality, multiple calculations were conducted, moving the critical assembly inside the cell and detection location outside the cell (see Figure 6). Increased dose due to streaming around the edges of the window assembly was not observed. When both the critical assembly and the detection point are near the center of the window, the dose-in-air is maximized. With the criticality source located 2 ft from the cell wall and the detection point on the outer surface of the window, the calculated dose was $2.7 \pm 2.2\%$ Rad-in-air. With no window in place, the dose was calculated to be 21,000 rem.

To determine what minimally constitutes a window, models were evaluated where the mineral oil, the outer most glass slab (J), and the inner most glass slab (A) were all omitted. This was done to determine operability requirements for the window and to allow maintenance on oil and protective slabs. In the situation where all three of these window elements are omitted, the dose through remaining window is comparable to the intact window.
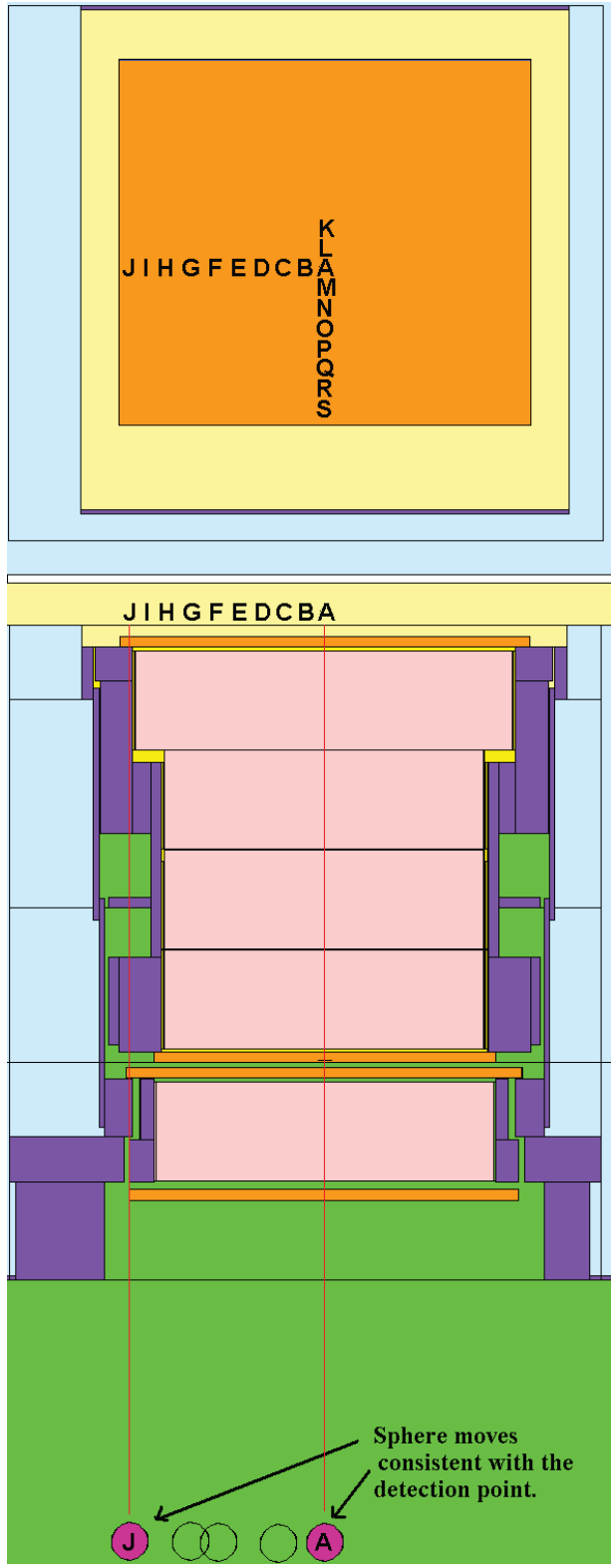
Figure 6. Determining the Maximum Dose Location.

Calculations were performed with various changes to the shielding to determine the direct radiation shielding performance for the fission product source. In general, each calculation required 200,000,000 source particles and 2,300 minutes to complete. No advanced variance reduction techniques were applied. Results of the calculations are shown in Table 2. Except where noted, the detector location is 1 cm away from the center of the window face. As noted above, the radiation source consisted of ^{137}Cs contained in the molten electrorefiner salt.

Table 2. Dose Rate Results.

Configuration	Dose Result [rem/hr]
No shielding	$2.3 \times 10^3 \pm 1.1\%$
Cell walls in place and all glass and oil removed	$1.0 \times 10^3 \pm 0.12\%$
Cell walls in place, glass in place except A and J slab, and oil missing	$6.2 \times 10^{-13} \pm 5.9\%$
Cell walls and glass in place and all oil removed	$2.4 \times 10^{-13} \pm 8.0\%$
All shielding in place including glass and oil	$2.0 \times 10^{-13} \pm 6.8\%$
All shielding in place with detector location in front of concrete rather than glass	$4.2 \times 10^{-15} \pm 4.3\%$

The "no shielding" results indicate the obvious need for shielding. Results with shielding show the outstanding shielding provided by the cell walls and windows. In particular, the results also indicate removal of the oil, the A-slab, and the J-slab presents no appreciable increase in the radiation levels outside the cell. The last entry in the table indicates the concrete shielding is approximately two orders of magnitude more effective than the window.

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1. LA-UR-03-1987, "MCNP – A General Monte Carlo N-Particle Transport Code, Version 5."