Document ID: EDF-6291
Revision ID: 1
Effective Date: 09/12/06

Engineering Design File

Project No.

Radiological Characterization of the Engineering Test Reactor (ETR) Complex Internal Surfaces

Idaho Cleanup Project

The Idaho Cleanup Project is operated for the
U.S. Department of Energy by CH2M • WG Idaho, LLC

ENGINEERING DESIGN FILE

EDF No.: 6291 EDF Rev. No.: 1 Project File No.:

1. Title: Radiological Characterization of the Engineering Test Reactor (ETR) Complex Internal Surfaces

2. Index Codes:
Building/Type NA SSC ID NA Site Area RTC

3. NPH Performance Category: ______ or ☒ N/A

4. EDF Safety Category: ______ or ☒ N/A SCC Safety Category: ______ or ☒ N/A

5. **Summary:**
This EDF a radiological characterization of the Engineering Test Reactor (ETR) complex internal surfaces (the surfaces inside piping, equipment, tanks, and the reactor). This characterization is necessary for establishing radiological work controls and source term calculations.

6. Review (R) and Approval (A) and Acceptance (Ac) Signatures:
(See instructions for definitions of terms and significance of signatures.)

	R/A	Typed Name/Organization	Signature	Date
Performer/ Author	N/A	Craig A. Nesshoefer/BSI	[signature]	8/23/06
Technical Checker	R	Dean Stewart/BSI	Per e-mail	8/24/06
Independent Peer Reviewer (if applicable)				
Approver	A	W. R. Spruill/CWI 7410	[signature]	8/28/06
Approver	A	A. L. Nellesen/CWI 7410	[signature]	8/29/06
Requestor (if applicable)	Ac	Bruce Culp/PS2	Per e-mail	8/23/06
Doc. Control		Becky Metcalf	Becky Metcalf	9/12/06

7. Distribution: (Name and Mail Stop)

8. Does document contain sensitive unclassified information? ☐ Yes ☒ No
If Yes, what category:

9. Can document be externally distributed? ☒ Yes ~~☒ No~~

10. Uniform File Code: 0250 Disposition Authority: Envi-b-4-a
Record Retention Period: Cut off when superseded, obsolete or canceled. Destroy 75 years after cutoff

11. For QA Records Classification Only: ☐ Lifetime ☐ Nonpermanent ☐ Permanent
Item and activity to which the QA Record apply: NA

12. NRC related? ☐ Yes ☒ No

EDF No.: 6291 EDF Rev. No.: 1 Project File No.:

1. Title:	Radiological Characterization of the Engineering Test Reactor (ETR) Complex Internal Surfaces
2. Index Codes:	Building/Type NA SSC ID NA Site Area RTC
13. Registered Professional Engineer's Stamp (if required)	N/A

1. PURPOSE

This EDF a radiological characterization of the Engineering Test Reactor (ETR) complex internal surfaces (the surfaces inside piping, equipment, tanks, and the reactor). This characterization is necessary for establishing radiological work controls and source term calculations.

2. INTRODUCTION

Characterization of the interior surfaces of piping and components will be used for the following:

- Establish required work controls such as PPE requirements, survey requirements, and air sampling requirements.
- Determine source terms for waste containers.
- Calculation of the source term for the internal surfaces of the ETR facility.

The internal surface source term will be used for the determination of activity of the ETR complex at various grade levels and in the determination of the total source term of the ETR complex.

3. ANALYTICAL DATA

In September of 2005 a sample (pipe scale) was taken from the primary system at a heat exchanger low point drain. The sample consisted of activated corrosion products and was sent to an off-site lab for analysis.

The lab results are presented in TRA77201RN (reference 1). Only data that was statistically significant (greater than the 1 sigma uncertainty of the analysis) and greater than the minimum detectable activity were used in this analysis. A summary of the results and their Co-60 normalized values are shown in Table 1 below.

Table 1. Lab Results and Ratio to Co-60

Nuclide	pCi/g	Normalized to Co60
Co60	2.61E+06	1.00E+00
Cs137	1.87E+04	7.16E-03
C14	1.44E+03	5.52E-04
Fe55	3.55E+06	1.36E+00
Ni63	6.38E+06	2.44E+00
Ni59	6.69E+04	2.56E-02
Sr90	1.75E+04	6.70E-03
Am241	8.99E+02	3.44E-04
Pu238	1.03E+03	3.95E-04
Pu239	1.60E+03	6.13E-04
U233	1.57E+02	6.02E-05
U235	4.21E+00	1.61E-06

4. METHODS AND ASSUMPTIONS

The assumptions made for this analysis are as follows:

- The relative ratios between the radionuclides are the same on the internal surfaces as they are in the corrosion products that were sampled. This assumption is considered valid for the following reasons:

    a. The sample point was a low point drain from a primary heat exchange where impurities in the primary system water would have collected and waste drain lines would have been contaminated from the primary water.

    b. The sample results above contained radionuclides associated with fission/fuel (Cs-137, Sr-90, Am-241, Pu-238, Pu-239, U-233, and U-235) and radionuclides associated with activation (Co-60, C-14, Fe-55, Ni-59. Ni-63) indicating that the contamination came from or passed through the reactor and were then deposited on piping surfaces.

- The contamination is generally uniformly distributed throughout these contaminated systems surfaces.

    a. During reactor operation, the same water flowed through the reactor as the primary system.

    b. The primary system did contain an ion exchanger used to remove activity from the primary coolant. However, less than 1% of primary system flow was through the ion exchanger so it is assumed that contamination levels upstream and downstream of the ion exchanger are the same

A MicroShield (reference 2) model was created to determine the levels of internal contamination that would result in a contact dose rate equivalent to a measured contact dose rate. Radiological surveys were performed throughout the ETR complex on the primary and waste drain piping. The dose rate results documented on these surveys (reference 3) showed radiation levels of less than 0.5 mrem/hr on contact with the exterior of the pipe for the vast majority of these piping systems, a contact dose rate of 0.5 mrem/hr was used in the model. The model was made with the following parameters:

- The piping is modeled as a cylindrical surface (representing the pipe surface contamination)

- Piping diameter is 24” (this size of pipe has the most total surface area in the ETR primary system).

- The pipe has a 0.38” wall thickness.

- The pipe was modeled with a 120” length. The primary pipe was not cut into sections when the surveys were performed so 120” is used to allow the model to conservatively account for radiations emitted from contamination distant from the actual survey location.

MicroShield models were also created for a 5’ length, a 1’ length, and 36” diameter pipe. The results of the 24” diameter pipe were more conservative (higher source term value) than the 36” pipe models and the models for the decreased pipe lengths did not significant change the activity per unit area values.

The MicroShield results (in Ci) were then divided by the surface area of the pipe section to give an activity per unit area value in Ci/cm^2 for each radionuclide shown in Table 2 below. The activity per unit area values are multiplied by the surface area for each piping system (Appendix A, provided by ETR Engineering) to obtain a total activity for each piping system. Results of these calculations are presented in Table 3.

5. RESULTS

Table 2. Summary of the MicroShield results and the activity per unit area values.

Nuclide	24" OD 10' length (Ci)	24" OD (Ci/cm^2)
Am241	6.24E-08	1.1E-12
C14	1.00E-07	1.8E-12
Co60	1.81E-04	3.2E-09
Cs137	1.30E-06	2.3E-11
Fe55	2.47E-04	4.4E-09
Ni59	4.64E-06	8.2E-11
Ni63	4.42E-04	7.8E-09
Pu238	7.16E-08	1.3E-12
Pu239	1.11E-07	2.0E-12
Sr90	1.21E-06	2.1E-11
U233	1.09E-08	1.9E-13
U235	2.92E-10	5.2E-15

Table 3. Summary of Internal Surface activity for ETR by System and Nuclide.

Area Description	Surface Area (cm^2)	Am241 (Ci)	C14 (Ci)	Co60 (Ci)	Cs137 (Ci)	Fe55 (Ci)	Ni59 (Ci)	Ni63 (Ci)	Pu238 (Ci)	Pu239 (Ci)	Sr90 (Ci)	U233 (Ci)	U235 (Ci)
Reactor Vessel	1520713	1.68E-06	2.69E-06	4.88E-03	3.49E-05	6.63E-03	1.25E-04	1.19E-02	1.93E-06	2.99E-06	3.27E-05	2.94E-07	7.85E-09
Nozzle Trench	148645	1.64E-07	2.63E-07	4.77E-04	3.41E-06	6.48E-04	1.22E-05	1.16E-03	1.88E-07	2.92E-07	3.19E-06	2.87E-08	7.67E-10
Canal	957273	1.06E-06	1.69E-06	3.07E-03	2.20E-05	4.17E-03	7.86E-05	7.49E-03	1.21E-06	1.88E-06	2.06E-05	1.85E-07	4.94E-09
Console Floor	48619	5.36E-08	8.61E-08	1.56E-04	1.12E-06	2.12E-04	3.99E-06	3.80E-04	6.16E-08	9.56E-08	1.04E-06	9.39E-09	2.51E-10
Reactor Basement	3071252	3.39E-06	5.44E-06	9.85E-03	7.05E-05	1.34E-02	2.52E-04	2.40E-02	3.89E-06	6.04E-06	6.60E-05	5.93E-07	1.59E-08
Waste Tank Pits	1365888	1.51E-06	2.42E-06	4.38E-03	3.14E-05	5.96E-03	1.12E-04	1.07E-02	1.73E-06	2.68E-06	2.93E-05	2.64E-07	7.05E-09
AGS Cubicle	138240	1.52E-07	2.45E-07	4.43E-04	3.17E-06	6.03E-04	1.13E-05	1.08E-03	1.75E-07	2.72E-07	2.97E-06	2.67E-08	7.14E-10
M-3 / P-7 Cubicles	3280307	3.62E-06	5.81E-06	1.05E-02	7.53E-05	1.43E-02	2.69E-04	2.57E-02	4.15E-06	6.45E-06	7.05E-05	6.33E-07	1.69E-08
C-7 / M-13 / N-14 Cubicles	1797575	1.98E-06	3.18E-06	5.76E-03	4.13E-05	7.84E-03	1.48E-04	1.41E-02	2.28E-06	3.53E-06	3.86E-05	3.47E-07	9.28E-09
F-10 / H-10 Cubicles	1014026	1.12E-06	1.79E-06	3.25E-03	2.33E-05	4.42E-03	8.32E-05	7.93E-03	1.28E-06	1.99E-06	2.18E-05	1.96E-07	5.24E-09
L-12 / M-7	3558100	3.92E-06	6.30E-06	1.14E-02	8.17E-05	1.55E-02	2.92E-04	2.78E-02	4.51E-06	6.99E-06	7.64E-05	6.87E-07	1.84E-08
C-13 / G-16	1701364	1.88E-06	3.01E-06	5.46E-03	3.91E-05	7.42E-03	1.40E-04	1.33E-02	2.15E-06	3.34E-06	3.66E-05	3.28E-07	8.78E-09
Control Rod Access Room	267561	2.95E-07	4.74E-07	8.58E-04	6.14E-06	1.17E-03	2.20E-05	2.09E-03	3.39E-07	5.26E-07	5.75E-06	5.16E-08	1.38E-09
Subpile	1736098	1.92E-06	3.07E-06	5.57E-03	3.99E-05	7.57E-03	1.43E-04	1.36E-02	2.20E-06	3.41E-06	3.73E-05	3.35E-07	8.96E-09
GEEL Tunnel & Delay Tanks	12938049	1.43E-05	2.29E-05	4.15E-02	2.97E-04	5.64E-02	1.06E-03	1.01E-01	1.64E-05	2.54E-05	2.78E-04	2.50E-06	6.68E-08
PCS Piping Tunnel	4599320	5.07E-06	8.14E-06	1.47E-02	1.06E-04	2.01E-02	3.78E-04	3.60E-02	5.83E-06	9.04E-06	9.88E-05	8.88E-07	2.37E-08
HX Bldg Bypass Demin Tank and Valve Room	2571038	2.84E-06	4.55E-06	8.24E-03	5.90E-05	1.12E-02	2.11E-04	2.01E-02	3.26E-06	5.05E-06	5.52E-05	4.96E-07	1.33E-08
HX Bldg Main Floor and Lower Level	79409802	8.76E-05	1.41E-04	2.55E-01	1.82E-03	3.46E-01	6.52E-03	6.21E-01	1.01E-04	1.56E-04	1.71E-03	1.53E-05	4.10E-07
Degassing Room	1239698	1.37E-06	2.19E-06	3.98E-03	2.85E-05	5.41E-03	1.02E-04	9.70E-03	1.57E-06	2.44E-06	2.66E-05	2.39E-07	6.40E-09
Pump Pits	3032355	3.34E-06	5.37E-06	9.72E-03	6.96E-05	1.32E-02	2.49E-04	2.37E-02	3.84E-06	5.96E-06	6.51E-05	5.85E-07	1.57E-08
Cubicle Exhaust Booster Blower Room	1144491	1.26E-06	2.03E-06	3.67E-03	2.63E-05	4.99E-03	9.40E-05	8.95E-03	1.45E-06	2.25E-06	2.46E-05	2.21E-07	5.91E-09
	Total (Ci)=	1.38E-04	2.22E-04	4.03E-01	2.88E-03	5.47E-01	1.03E-02	9.82E-01	1.59E-04	2.47E-04	2.70E-03	2.42E-05	6.48E-07
											Total Interior Ci =		1.95E+00

6. VERIFICATION AND VALIDATION OF MICROSHIELD V6.10

MicroShield v6.10 (reference 2) was used to develop a model of the geometry and material composition of the waste. MicroShield is maintained under INL Software Configuration and Control Number 161146 and is fully validated, verified, and controlled in accordance with applicable INL requirements. MicroShield is installed on CPU number 382434, which is located at TRA-1601.

7. SOURCES OF ERROR/UNCERTAINTY ANALYSIS

Sources of uncertainty in the characterization of the ETR complex internal surfaces consist of the following:

- Uncertainty in the in the lab analysis performed on the sample used to identify the radionuclides in the contamination and the uncertainty associated with scaling to Co-60
- Uncertainty associated with the distribution of the radionuclides in the contamination
- Uncertainty with the MicroShield model
- Measurement uncertainties associated with the dose rate measurements
- Uncertainty associated with the contaminated surface area.

1. Lab Analysis

The TRA77201RN lab results, associated uncertainties, and the % error (determined by the equation 100 X error/reported value) for the sample removed from the primary system are shown in the Table 3 below. The lab analysis was performed at an off-site laboratory and estimated values for the uncertainty associated with the detector geometry and sample geometry could not be located on the lab report. Similar laboratory analyses were performed by the Test Reactor Radioanalytical Laboratory when analyzing swipe samples used to characterize the ETR external surfaces. The uncertainty associated with detector geometry and sampling geometry were both 5% when analyzing the swipe samples and this same uncertainty is assumed for the lab data below. Uncertainties are propagated in quadrature and expressed at one standard deviation

Table 4. Laboratory Data.

Radionuclide	Sample Activity (pCi/g)	Sample Uncertainty (+/-)	Propagated % Error (+/-)
Am-241	8.99E+02	1.09E+02	14.04
Pu-238	1.03E+03	9.77E+01	11.83
Pu239/240	1.60E+03	1.34E+02	10.96
U233/234	1.57E+02	1.29E+01	10.84
U235	4.21E+00	1.74E+00	41.93
Co-60	2.61E+06	2.83E+04	7.15
Cs-137	1.87E+04	2.69E+03	16.03
C-14	1.44E+03	3.30E+01	7.43
Fe-55	3.55E+06	8.90E+04	7.50
Ni-63	6.38E+06	1.08E+05	7.27
Ni-59	6.69E+04	2.14E+03	7.76
Sr-90	1.75E+04	2.47E+02	7.21

From the above data, Co-60, Fe-55, and Ni-63 are the dominant radionuclides and represent ~99% of the activity in the sample. Co-60 is the dominant gamma emitting radionuclide and is the nuclide to which the other radionuclides were scaled to for use in the MicroShield model. To scale a radionuclide to Co-60, the activity of that radionuclide is divided by activity of Co-60. The equation given in the MARSSIM for propagating uncertainty in this division operation is shown below:

$$\sigma_u = u\sqrt{\left(\frac{\sigma_x}{x}\right)^2 + \left(\frac{\sigma_y}{y}\right)^2}$$

Where:

σ_u = propagated uncertainty

u = is the value determined by dividing the measured values x and y

x & y = measurement values

σ_x = standard deviation or uncertainties associated with measurement x

σ_y = standard deviation or uncertainties associated with measurement y

The results of this calculation are shown in Table 4 below.

Table 5. Uncertainty associated with Scaling.

Nuclide Scaled to Co-60	Nuclide lab Result "X" (pCi/g)	Co-60 Lab Result “Y” (pCi/g)	u	σ_x	σ_y	σ_u	% Uncertainty
Fe-55	3.55E+06	2.61E+06	1.36E+00	2.66E+05	1.87E+05	1.41E-01	10.37%
Ni-63	6.38E+06	2.61E+06	2.44E+00	4.64E+05	1.87E+05	2.49E-01	10.20%

From this analysis, 10.37% will be used as the total propagated uncertainty associated with the lab analysis and the subsequent scaling.

2. Distribution of Radionuclides in the Contamination

During the characterization of the ETR internal surfaces, no attempt was made to determine the variations in the relative ratios between the various radionuclides from point to point in the contaminated systems. As stated previously, the sample chosen was assumed to be representative of the internal surfaces for the following reasons:

- The sample results showed both nuclides associated with activation (Co-60, Ni-63, etc.) and with fission/fuel (Cs-137, Pu-239, etc.)
- The sample was performed at a low-flow area where contaminates in the primary system would tend to concentrate
- While the primary system did contain a demineralizer used for removal of contaminants from the primary system, the sample was removed from the system in a location between the reactor outlet and the demineralizer. Additionally, less than 1% of the coolant flow was through the demineralizer so it is assumed that the demineralizer made an insignificant change in the upstream and downstream contamination levels
- During reactor operation, water continuously flowed through the reactor. This water distributed the contamination throughout the primary system.

3. Uncertainty Associated with the MicroShield Model

Activity values associated with the primary system were based on a conservative model. 24” and 36” diameter pipes are the most abundant (in terms of total surface area) sizes in the primary system. The model for the 24” pipe size produced a higher area specific activity value (Ci/cm^2) than did the model for the 36” pipe so the area specific activity values for the 24” pipe were used. Additionally, the results from detailed surveys of the primary piping showed that the vast majority of the piping had radiation levels documented as less than 0.5 mrem/hr. In the model 0.5 mrem/hr was used as the modeled dose rate to determine the area specific activity values for the piping. Hot spots did exist in the primary system however these spots were small and have since been removed. While an uncertainty value was not determined for the model, the results are thought to be conservative.

4. Measurement uncertainties associated with the dose rate measurements

   Per Byron H. Christiansen of the INL Health Physics Instrumentation Laboratory, INL dose rate instruments are calibrated to +/-10% of the conventionally true value. The stated uncertainty of field measurements is +/-20% based on field instrument performance check which allows +/-20% variance in the instrument readings before they are considered to be out of tolerance.

   As stated previously, the vast majority of the piping systems were documented as being less than 0.5 mrem/hr and 0.5 mrem/hr was conservatively used in the model. For this uncertainty evaluation, since it cannot be determined from the survey documentation how far below 0.5 mrem/hr the actual dose rate is, it is assumed that the dose rate is only slightly less than 0.5 mrem/hr. With this assumption, the uncertainty associated with the dose rate measurement is +/-20%

5. Uncertainty associated with the contaminated surface area.

   No attempt was made during the characterization of the internal surfaces to determine an uncertainty associated with the estimate that ETR engineering made in the internal piping surfaces. Per conversation with the engineer that developed the data, the estimated surface areas are thought to be conservative (more surface area is estimated than actually exists). This conservative estimate of the surface area results in a conservative source term since the source term is directly related to the amount of contaminated surface area

Propagated Error for the Internal Surfaces (calculated in quadrature)

The total propagated error associated with the determination of the internal surface source term is presented in Table 6 below and is a result of uncertainty associated with the sampling/scaling and measurement error.

Table 6. Total Propagated Uncertainty.

Source of Error	Percent Uncertainty
Sampling/Scaling	10.37%
Measurements	20%
Radionuclide Distribution	-
Model	-
Contaminated Surface Area	-
Total Propagated Error =	+/-23%

The calculated source term is thought, however, to be conservative. In addition to the conservative assumptions and calculations discussed above, actual surveys performed on the internal surfaces confirm the conservatism of the calculations.

Based on the 0.5 mR/hr dose rate, the internal surface contamination in the primary system was calculated to be 3.45E06 dpm/100 cm^2. This is the level of loose surface contamination that is also assumed to exist in the reactor.

The vast majority of the contamination surveys performed on the primary system internals during strip-out activities were documented as being less than 1.0E05 dpm/100 cm^2. Additionally, process history indicates that after the reactor was drained, it was flushed with water to reduce contamination levels.

This lower level of contamination on the vessel interior surfaces is demonstrated by the contamination levels on equipment removed from the reactor during the Beryllium sampling effort inside of the reactor were a maximum of 2,500 dpm/100 cm^2.

8. CONCLUSION

This EDF documents the radiological characterization of the Engineering Test Reactor (ETR) complex internal surfaces (the surface inside piping, equipment, tanks, and the reactor) and determines source terms for these surfaces.

This data will also be used to establish work control requirements and in the determination of the ETR complex source term.

9. REFERENCES

1. TRA BUILDINGS AND STRUCTURES, SDG Number TRA77201RN

2. MicroShield v6.10 Shielding Code, Grove Engineering

3. ETR Radiological Survey Log, 2005.

10. APPENDIXES

Appendix A, Engineering Provided Surface Areas

Appendix B, MicroShield Results

Appendix A

Engineering Provided Surface Areas

Appendix A

Piping Surface Areas

ETR Contaminated Piping Inventory

Area Description	Volume (ft^3)	Estimated Piping Length (ft)	Surface Area (ft^2)	Elevation	Notes
Reactor Vessel	4914	N/A	1637	1	
Nozzle Trench	10	204	160	1	Assume 3" Average Diameter
Canal	64	1313	1030	4	Assume 3" Average Diameter
Console Floor	2	100	52	3	Assume 2" Average Diameter Warm Waste Feed From ETR-653
Reactor Basement (General Area)	1240	702	3306	4	Assume 18" Average Diameter
Waste Tank Pits	1343	380	1470	4	Assume 4" Average Pipe Diameter. Includes Warm and Cold (Decon) Waste Tanks
AGS Cubicle	3.1	569	149	4	Assume 1" Average Diameter
M-3 / P-7 Cubicles	831.17	1823	3531	4	Assume 4" Average Pipe Diameter and 18" Tank and Valve Diameter. Includes Pressurizers.
J-10 / L-10 Cubicles	N/A	N/A	N/A	4	
C-7 / M-13 / N-14 Cubicles	800	596	1935	4	Assume 2" Average Diameter
F-10 / H-10 Cubicles	657.33	2732	1091	4	Assume 4" Average Pipe Diameter and 18" Tank and Valve Diameter. Includes Surge Tank
L-12 / M-7	1469	2182	3830	4	Assume 4" Average Pipe Diameter and 18" Tank and Valve Diameter. Includes Pressurizers.
C-13 / G-16	651.5	120	1831	4	Assume 4" Average Pipe Diameter and 18" Tank and Valve Diameter.
Helium System Cubicles	N/A	N/A	N/A	4	
SLSF FS&R	N/A	N/A	N/A	4	
Control Rod Access Room	6	1101	288	4	Assume 1" Average Diameter
Subpile	119	1334	1869	4	Assume 4" Average Pipe Diameter and 18" Cooler Diameter.
GEEL Tunnel & Delay Tanks	16145	3823	13926	4	Assume 8" Average Diameter. Includes Delay Tanks
PCS Piping Tunnel	3713	267	4951	3,4	Assume 36" Pipe Diameter and 36" Valve Diameter
Compressor Building Main Room	N/A	N/A	N/A	2	
HX Bldg Bypass Demin Tank and Valve Room	1121	1796	2767	1	Assumes 4" Pipe Diameter. Resin Tanks Included.
HX Bldg Main Floor and Lower Level	10652	1760	12358	2,3,4	Assumes 24" Pipe Diameter. Heat Exchangers Included.
Degassing Room	1256	1590	1334	1	Assumes 2" Pipe Includes Degassing Tank
Secondary Pipe Trench	N/A	N/A	N/A	2,3,4	
Pump Pits	2064	413	3264	1	Assumes 24" Pipe Diameter.
Cubicle Exhaust Booster Blower Room	844	154	1232	1	Assumes 24" Pipe Diameter.

1 = Above Grade
2 = 0 - 4'
3 = 0 - 10'
4 = 10' and below

Appendix B

MicroShield Results

Appendix B

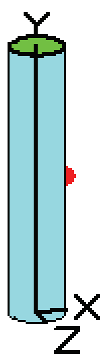
MicroShield Results

MicroShield v6.10 (0063)
INL

Page	1
DOS File	Primary 24 x 10.ms6
Run Date	November 17, 2005
Run Time	2:52:04 PM
Duration	00:00:00

File Ref
Date
By
Checked

Case Title: 24" Primary
Description: 10 length of pipe
Geometry: 10 - Cylinder Surface - External Dose Point



Source Dimensions:

Height	304.8 cm	(10 ft 0.0 in)
Radius	29.515 cm	(11.6 in)

Dose Points

A	X	Y	Z
# 1	33.02 cm	152.4 cm	0 cm
	1 ft 1.0 in	5 ft 0.0 in	0.0 in

Shields

Shield N	Dimension	Material	Density
Cyl. Radius	11.62 in	Air	0.00122
Transition		Air	0.00122
Air Gap		Air	0.00122
Wall Clad	.38 in	Iron	8

Source Input : Grouping Method - Linear Energy
Number of Groups : 25
Lower Energy Cutoff : 0.015
Photons < 0.015 : Included
Library : Grove

Nuclide	curies	becquerels	μCi/cm²	Bq/cm²
Am-241	6.2351e-008	2.3070e+003	1.1031e-006	4.0814e-002
Ba-137m	1.2277e-006	4.5425e+004	2.1720e-005	8.0363e-001
C-14	1.0005e-007	3.7019e+003	1.7701e-006	6.5492e-002
Co-60	1.8125e-004	6.7063e+006	3.2066e-003	1.1865e+002
Cs-137	1.2978e-006	4.8017e+004	2.2960e-005	8.4950e-001
Fe-55	2.4650e-004	9.1206e+006	4.3610e-003	1.6136e+002
Ni-59	4.6401e-006	1.7168e+005	8.2090e-005	3.0373e+000
Ni-63	4.4226e-004	1.6363e+007	7.8242e-003	2.8950e+002
Pu-238	7.1595e-008	2.6490e+003	1.2666e-006	4.6865e-002
Pu-239	1.1111e-007	4.1110e+003	1.9657e-006	7.2730e-002
Sr-90	1.2144e-006	4.4933e+004	2.1484e-005	7.9493e-001
U-233	1.0911e-008	4.0372e+002	1.9304e-007	7.1425e-003
U-235	2.9182e-010	1.0797e+001	5.1627e-009	1.9102e-004
Y-90	1.2144e-006	4.4933e+004	2.1484e-005	7.9493e-001

Buildup : The material reference is - Wall Clad
Integration Parameters

Y Direction (axial)	20
Circumferential	20

Results

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup
0.0328	3.365e+03	4.363e-26	4.883e-26	3.332e-28	3.728e-28
0.0596	8.346e+02	1.006e-08	1.423e-08	2.015e-11	2.850e-11
0.1219	4.026e+00	4.707e-07	9.466e-07	7.379e-10	1.484e-09
0.1859	7.160e+00	3.706e-06	8.616e-06	6.434e-09	1.496e-08
0.2214	1.080e-02	8.363e-09	2.001e-08	1.508e-11	3.609e-11
0.6616	4.087e+04	1.983e-01	4.040e-01	3.844e-04	7.832e-04
0.6938	1.094e+03	5.698e-03	1.146e-02	1.100e-05	2.212e-05
1.1732	6.706e+06	7.576e+01	1.314e+02	1.354e-01	2.348e-01
1.3325	6.706e+06	9.098e+01	1.524e+02	1.578e-01	2.644e-01
Totals	**1.346e+07**	**1.669e+02**	**2.842e+02**	**2.936e-01**	**5.000e-01**