PIQUA NUCLEAR POWER FACILITY
ENTOMBMENT EXPOSURE EVALUATION – FINAL REPORT

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EXECUTIVE SUMMARY

The Piqua Nuclear Power Facility was a demonstration reactor under the Atomic Energy Commission. It was decommissioned in 1966. The facility has been defueled and the entire reactor complex has been entombed in steel reinforced concrete.

The United States Department of Energy (DOE) currently provides regulatory oversight for the facility which is owned by the City of Piqua. The DOE and the Army Corps of Engineers have evaluated removing all above ground structures, applying an additional layer of concrete to entombment, and allowing the City of Piqua to use the site as a parking lot/laydown yard.

As described in this report, it will take approximately 100 years for the radioactive material in the entombment to decay to levels where there is no potential external exposure hazard to members of the public. It will take a little over one million years for the longest lived radionuclides to decay to levels below current screening guidelines. However, the potential exposure to these long lived isotopes is considered to be an extremely unlikely event.

The conclusion reached in this report is that any further remediation of the entombment other than adding the additional concrete is not warranted and that current controls are adequate to minimize public exposure to a negligible level.

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1.0 PURPOSE

This report describes the technical basis for the radiological dose modeling of the former Piqua Nuclear Power Facility (PNPF) once demolished, and evaluates the results of the modeling against current exposure standards. The exposure modeling is used by regulatory agencies to determine the risk to the public from leaving the entombment in place. The modeling results also identify the length of time that government control over the subsurface, entombed radiological material would be necessary to ensure the protection of human health and the environment.

2.0 SCOPE

This report covers the technical basis for the modeling efforts chosen and evaluates other pathways that were considered but not evaluated. The estimated exposure from the entombment is documented along with all assumptions used in developing the estimate. The results are compared to direct survey results where possible.

This report does not evaluate an exposure estimate for entombment removal. Such an evaluation would require a work plan for removal which has not been developed and is considered outside the scope of this evaluation.

3.0 REGULATORY BASIS

3.1 DEPARTMENT OF ENERGY

In 10 CFR 835, *Occupational Radiation Protection*, the DOE sets standards for protection of workers and members of the public. 10CFR835.208 sets a limit of 100 mrem per year for members of the public entering a controlled area. There is no expectation of establishing any radiological controls to restrict access top of the entombment. 10CFR835.202 sets an occupational exposure limit of 5 rem per year which is 50 times higher than the public dose limit. The DOE further restricts this administratively to 500 mrem per year as an administrative level requiring management approval to exceed. 10CFR835.203 requires the summing of internal and external dose to determine compliance with the limits. This is the Total Effective Dose (TED) equivalent and is the method used by ANSI N13.12-2013 to determine screening levels for volumetric contamination. Throughout 10CFR835, the DOE emphasizes the use of the As Low As is Reasonably Achievable (ALARA) process to minimize personnel exposure.

DOE Order 458.1. *Radiation Protection of the Public and the Environment*, sets objectives to conduct activities to maintain exposure to members of the public within limits it has established, to provide controls for clearance of property, and to keep exposures ALARA. A public dose limit of 100 mrem/y is specified. Demonstration that exposure is ALARA is required. Where exposure from DOE sources is expected to be less than 25 mrem/y, no additional pathways are required to be evaluated. The order also requires protection of drinking water and groundwater. Release and clearance of property has a TED limit of 25 mrem/y. There is a requirement for public notice of intent to clear real property. This will need to be coordinated through the State of Ohio.

3.2 STATE OF OHIO

In rule 3701:1-38-22, *Decommissioning*, the State of Ohio Department of Health sets a limit of 25 mrem/y which is consistent with DOE guidance. This rule also requires protection of the drinking water and groundwater.

3.3 AMERICAN NATIONAL STANDARD INSTITUTE N13.12-2013

In March 2021, the DOE Office of the Associate Under Secretary for Environment, Health, Safety, and Security (AU) approved the use the American National Standard Institute (ANSI)/Health Physics Society (HPS) N13.12-2013, *Surface and Volume Radioactive Standards for Clearance* screening levels as the pre-approved Authorized Limits for Release and Clearance of Volumetric Radioactivity of Personal Property. Although the entombed reactor remaining in the site's subsurface would be categorized as real property, one of the purposes of this evaluation is to establish the timeframe when the entombment would achieve a release from DOE control. A release from DOE control could conceivably allow the dismantlement of the entombment which contains activated metals and concrete. Since this material could be recycled or used for other purposes, this evaluation utilizes the personal property standard as a conservative measure.

4.0 FACILITY DESCRIPTION

Between 1967 and 1969, the Atomic Energy Commission (AEC), predecessor to the DOE, decommissioned the PNPF and removed the reactor fuel, coolant, and most of the radioactive materials from the site. Contaminated piping and equipment inside the reactor building were removed or decontaminated. The reactor vessel, concrete biological shield (bioshield), and non-removable parts of the vessel were left in place. The entire reactor complex was entombed in two feet of steel reinforced concrete.

The PNPF is located in southwestern Ohio on the east bank of the Miami River in Miami County. The reactor is located at the southeastern edge of the City of Piqua on property owned by the city. A map of the Piqua area is shown in Figure II-1, and an aerial view of the site is shown in Figure II-2.

The reactor building is approximately 120 feet from the river's edge. Access to the site is over an improved road which also provides access to the Piqua Sewage Treatment Plant that is located just south of the reactor plant.

The facility consists of two buildings: Reactor Building and Auxiliary Building. The reactor building houses the reactor, fuel storage and handling facilities, the main heat transfer system, and the degasification system. The auxiliary building houses the control room, service areas, auxiliary process systems, and administrative areas. The auxiliary building is directly connected to the reactor building. Additional historical details are presented in the Radiological Historical Site Assessment (January, 2020).

The Auxiliary Building and the aboveground portion of the reactor building are anticipated to be removed as part of the site's demolition.

<u>Reactor Building</u>

The reactor building is a vertical cylindrical carbon steel pressure vessel with 50 feet of its 112-foot length located below grade. The inner wall of the building above the 100-ft level is reinforced concrete, 18 in. thick, gradually tapering to 3 in. at the top of the dome. The exterior portion of the building above grade is insulated to reduce sun load. The building rests on a concrete pad. The exterior of the building below ground level is protected from contact with the soil by a bitumastic coating. In addition, a cathodic protection system, no longer functional, was employed to reduce the corrosive attack on the steel shell.

Reactor Complex (Entombment)

The reactor complex contains the radioactive materials that will remain on site. In general, the complex consists of the reactor vessel (including all internal components), the cavity liner, the reinforced concrete biological shield, and all other materials within and bounded by the biological shield. The entire complex is covered with a layer of reinforced concrete. This concrete protects the underlying complex and barriers and is the new floor of the reactor building. The portion of the reactor complex containing the radioactivity is shown in Figure II-3.

To prevent access to the complex, the lines from the reactor vessel to the process area were cut within the shield and welded shut. The reactor vessel was closed and access to the vessel was precluded by barriers which cannot be readily removed. The penetrations into the biological shield, such as shield cooling coils and instrument thimble, were closed so that access was prevented.

Some radioactive material was left on the PNPF site. To prevent personnel access to the radioactivity, the material was enclosed in the biological shield. The reactor vessel, thermal shield, grid plates, and support barrels were left in place. The reactor nozzles and piping that penetrate the concrete shield were cut off and welded shut. The sleeves in the concrete through which the piping passed were closed. The reactor vessel lid was bolted on the reactor vessel. The floor plug over the reactor vessel was tack welded in place to prevent access to the reactor vessel top flange. Small pipe lines for the shield cooling, neutron window cooling, and nitrogen purge of the reactor cavity were cut, and plugs were driven into and welded to the pipes.

The concrete surrounding and immediately adjacent to the reactor vessel also contains radioactive material. Personnel access to this material was prevented by the non-activated concrete located farther from the reactor vessel.

5.0 TECHNICAL BASIS FOR EXPOSURE EVALUATION

5.1 SOURCE TERM

The source term needed in order to calculate exposure is based on the radioactive material inventory described in the Piqua Nuclear Power Facility Safety Analysis Report, AI-AEC-Memo-1270, August 31, 1968. There are three source terms listed in the Safety Analysis Report (SAR): Main Source, Secondary Sources, and Concrete Source. The source term listed is as of January 1966 when the reactor was shut down for decommissioning. The time given for decay and safe access in the SAR is based on an external dose rate of 0.2 mrem/h and on the potential to achieve a maximum permissible body burden if the activity were ingested. These standards are no longer applicable. Based upon current standards, the date for unrestricted release calculated in 1968 was estimated to be the year 2106. This calculation is no longer valid by current standards. The non-occupational exposure limit is now set by DOE Order 458.1 at 25 mrem/y above background. ANSI N13.12-2013 establishes screening levels for volumetric contamination that will give a total effective dose equivalent of one mrem/y, which is less than the 25 mrem/y decommissioning criteria established by the DOE.

The activity present in the entombment as of February 2021 was calculated using the following equation:

$$A = A_0 \times e^{-\lambda t}$$

Where:

- A = Activity at any time – t
- A_0 = Initial activity at time 0, assumed to be January 1966
- λ = decay constant = 0.693 divided by the half-life
- t = decay time, 55 years 1 month

One endpoint was calculated for when the entombment would meet release criteria. ANSI N13.12-2013 establishes levels at which material will be less than one mrem/y TED. The use of the TED is consistent with the decommissioning criteria which uses a TED of 25 mrem/y. Since the reactor components are activated versus contaminated, the volumetric screening levels from the ANSI standard were used as described in Section3.3.

The time necessary for the activity as of February 2021 to decay to meet the ANSI Screening Levels is calculated by:

$$t = \left(\frac{-T_{1/2}}{0.693}\right) \times \ln\frac{A}{A_0}$$

Where:

t = time in years to reach the desired activity
$T_{1/2}$ = half-life of the isotope of interest
A = endpoint of interest (ANSI Screening Level)
A_0 = activity of the isotope as on February 2021

Table 1 shows the isotopes of interest, the half-life, the initial activity, and the activity as of February 2021 for the Main Source as described in Table V-2 of the SAR. The main source term includes components such as the core support plate. The total activity contained in the primary source was divided by the total mass of the primary components to give a pCi/g value for comparison the to the ANSI screening levels. From the Piqua SAR, Table V-1 shows the mass of the components. The total mass is 2.73×10^7 g.

Table 1: Piqua Reactor Main Source Term

Isotope	Half-life (y)	Activity January 1966 (Ci)	Activity February 2021 (Ci)	Activity February 2021 (µCi)	Activity February 2021 (pCi/g)
Beryllium-10	2.50×10^{6}	3.81×10^{-6}	3.81×10^{-6}	3.81	1.40×10^{-1}
Carbon-14	5730	5.76×10^{-5}	5.72×10^{-5}	57.2	2.10
Iron-55	2.6	1.15×10^{5}	4.84×10^{-2}	4.84×10^{4}	1.77×10^{3}
Cobalt-60	5.26	1.49×10^{3}	1.05	1.05×10^{6}	3.85×10^{4}

Table 2 shows the time for the Main Source to decay to the ANSI Screening Level.

Table 2: Decay times for Main Source to Reach ANSI Screening Level

Isotope	Half-life (y)	Activity February 2021 (µCi)	Activity February 2021 (pCi/g)	ANSI N13-12 Screening Levels (pCi/g)	Time to Reach ANSI N13.12 Screening Levels (y)
Beryllium-10	2.50×10^{6}	3.81	1.40×10^{-1}	300	0
Carbon-14	5730	57.2	2.10	30	0
Iron-55	2.6	4.84×10^{4}	1.77×10^{3}	30,000	0
Cobalt-60	5.26	1.05×10^{6}	3.85×10^{4}	3	7.18×10^{1}

Note that a value of zero (0) indicates that no additional decay time is needed for that isotope to reach the screening level.

This table indicates the Main Source term, with the exception of Co-60, meets the ANSI standard for clearance. The Co-60 activity will meet clearance levels in year 2093 (72 years from year 2021).

Table 3 shows the Secondary Source term as described in Table VI-4 of the SAR. The secondary source term is made up of miscellaneous hardware items remaining in the reactor. Table V1-1 gives the mass of the individual components in the secondary source term. Table VI-4 gives the isotopic content of each component. The mass and component activity were used to calculate the pCi/g values for the secondary source term.

Table 3: Piqua Reactor Secondary Source Term

Isotope	Half-life (y)	Activity January 1966 (Ci)	Activity February 2021 (Ci)	Activity February 2021 (µCi)	Activity February 2021 (pCi/g)
Carbon-14	5730	$1.80x10^{-5}$	$1.79x10^{-5}$	17.9	$4.08x10^{1}$
Sodium-22	2.62	$3.80x10^{-5}$	$1.79x10^{-11}$	$1.79x10^{-5}$	$4.09x10^{-5}$
Chlorine-36	$3.08x10^{5}$	$1.60x10^{-4}$	$1.60x10^{-4}$	160	$3.65x10^{2}$
Argon-39	269	$4.00x10^{-6}$	$3.47x10^{-6}$	3.47	7.93
Calcium-41	$8.00x10^{4}$	$4.40x10^{-4}$	$4.40x10^{-4}$	440	$1.00x10^{3}$
Iron-55	2.6	430	$1.81x10^{-4}$	181	$1.86x10^{3}$
Cobalt-60	5.26	43	$3.03x10^{-2}$	$3.03x10^{4}$	$3.12x10^{5}$
Nickel-59	$8.00x10^{4}$	0.049	$4.90x10^{-2}$	$4.90x10^{4}$	$5.03x10^{5}$
Nickel-63	92.0	7.80	5.15	$5.15Ex10^{6}$	$5.29x10^{7}$
Molybdenum-93	4,000	$1.00x10^{-6}$	$9.91x10^{-7}$	0.991	$7.28x10^{1}$
Silver-108m	438	0.012	$1.10x10^{-2}$	$1.10x10^{4}$	$1.43x10^{5}$

Table 4 shows the time needed for the Secondary Source to reach the ANSI Screening Level.

Table 4: Decay Times for Secondary Source to Reach ANSI Screening Level

Isotope	Half-life (y)	Activity February 2021 (µCi)	Activity February 2021 (pCi/g)	ANSI 13.12 Screening Level (pCi/g)	Time to Reach ANSI Screening Level (y)
Carbon-14	5730	17.9	$4.08x10^{1}$	30	$4.80x10^{3}$
Sodium-22	2.62	$1.79x10^{-5}$	$4.09x10^{-5}$	3	0
Chlorine-36	$3.08x10^{5}$	160	$3.65x10^{2}$	30	$1.23x10^{6}$
Argon-39	269	3.47	7.93	30	NA
Calcium-41	$8.00x10^{4}$	440	$1.00x10^{3}$	30000	$3.90x10^{5}$
Iron-55	2.6	181	$1.86x10^{3}$	30000	5.22
Cobalt-60	5.26	$3.03x10^{4}$	$3.12x10^{5}$	3	78.3
Nickel-59	$8.00x10^{4}$	$4.90x10^{4}$	$5.03x10^{5}$	3000	$7.15x10^{5}$
Nickel-63	92.0	$5.15x10^{6}$	$5.29x10^{7}$	3000	$1.55x10^{3}$
Molybdenum-93	4,000	0.991	$7.28x10^{1}$	300	0
Silver-108m	438	$1.10x10^{4}$	$1.43x10^{5}$	30000	$5.19x10^{3}$

Note that a value of zero indicates that no additional decay time is needed to reach the ANSI Screening Level for that isotope.

Table 5 shows the concrete source term. The SAR did not list total activity for the concrete, rather it listed the isotopes in terms of concentration at 25 years post shutdown. In addition, the activated portion of the concrete is limited to the first 15 centimeters of the bioshield. The concentration in μCi/cm³ was converted to pCi/g using a concrete density of 2.4 g/cm³.

Table 5: Piqua Reactor Concrete Source Term

Isotope	Half-life (y)	Total Activity January 1991 (Ci)	Activity February 2021(Ci)	Activity February 2021 (pCi/g)
Hydrogen-3	$1.20x10^{1}$	$1.09x10^{1}$	$4.51x10^{-1}$	$5.37x10^{4}$
Beryllium-10	$2.50x10^{6}$	$1.37x10^{-8}$	$1.36x10^{-8}$	$1.62x10^{-3}$
Carbon-14	$5.70x10^{3}$	$4.55x10^{-4}$	$4.52x10^{-4}$	$5.38x10^{1}$
Sodium-22	2.60	$4.55x10^{-6}$	$1.91x10^{-12}$	$2.28x10^{-7}$
Aluminum-26	$7.40x10^{5}$	$1.75x10^{-8}$	$1.75x10^{-8}$	$2.08x10^{-3}$
Chlorine-36	$3.00x10^{5}$	$1.12x10^{-3}$	$1.12x10^{-3}$	$1.33x10^{2}$
Argon-39	$2.70x10^{2}$	$9.10x10^{-2}$	$7.90x10^{-2}$	$9.41x10^{3}$
Potassium-40	$1.30x10^{9}$	$1.82x10^{-6}$	$1.82x10^{-6}$	$2.17x10^{-1}$
Calcium-41	$7.70x10^{4}$	$4.55Ex10^{-3}$	$4.55x10^{-3}$	$5.41x10^{2}$
Iron-55	2.70	$1.26x10^{-1}$	$9.13x10^{-8}$	$1.09x10^{-2}$
Cobalt-60	5.30	$1.93x10^{-1}$	$1.43x10^{-4}$	$1.71x10^{1}$
Palladium-107	$7.00x10^{6}$	$5.60x10^{-14}$	$5.60x10^{-14}$	$6.67x10^{-9}$
Silver-108m	$4.38x10^{2}$	$1.05x10^{-5}$	$9.62x10^{-6}$	1.15
Cadmium-109	1.30	$7.00x10^{-11}$	$1.24x10^{-23}$	$1.48x10^{-18}$
Cadmium-113	$1.40x10^{1}$	$1.30x10^{-6}$	$8.48x10^{-8}$	$1.01x10^{-2}$
Promethium-147	2.70	$5.25x10^{-11}$	$3.81x10^{-17}$	$4.53x10^{-12}$
Samarium-146	$1.20x10^{8}$	$1.93x10^{-12}$	$1.92x10^{-12}$	$2.29x10^{-7}$
Samarium-151	$9.00x10^{1}$	$4.20x10^{-3}$	$2.75x10^{-3}$	$3.27x10^{2}$
Europium-152	$1.20x10^{1}$	$4.90x10^{-1}$	$2.04x10^{-2}$	$2.42x10^{3}$
Europium-154	$1.60x10^{1}$	$2.24x10^{-2}$	$2.06x10^{-3}$	$2.45x10^{2}$
Europium-155	1.80	$1.16x10^{-11}$	$7.13x10^{-21}$	$8.49x10^{-16}$
Terbium-158	$1.50x10^{2}$	$2.14x10^{-12}$	$1.66x10^{-12}$	$1.97x10^{-7}$

Table 6 shows the decay times for the Concrete Source term to reach the ANSI Screening Level.

Table 6: Decay Time for Concrete Source Term to Reach ANSI Screening Level

Isotope	Half-life (y)	Activity February 2021 (Ci)	Activity February 2021 (pCi/g)	ANSI 13.12 Screening Level (pCi/g)	Time to Reach ANSI Screening Level (y)
Hydrogen-3	1.20×10^{1}	4.51×10^{-1}	5.37×10^{4}	3000	4.99×10^{1}
Beryllium-10	2.50×10^{6}	1.36×10^{-8}	1.62×10^{-3}	300	0
Carbon-14	5.70×10^{3}	4.52×10^{-4}	5.38×10^{1}	30	4.80×10^{3}
Sodium-22	2.60	1.91×10^{-12}	2.28×10^{-7}	3	0
Aluminum-26	7.40×10^{5}	1.75×10^{-8}	2.08×10^{-3}	300	0
Chlorine-36	3.00×10^{5}	1.12×10^{-3}	1.33×10^{2}	30	6.46×10^{5}
Argon-39	2.70×10^{2}	7.90×10^{-2}	9.41×10^{3}	30	2.24×10^{3}
Potassium-40	1.30×10^{9}	1.82×10^{-6}	2.17×10^{-1}	3	0
Calcium-41	7.70×10^{4}	4.55×10^{-3}	5.41×10^{2}	30000	0
Iron-55	2.70	9.13×10^{-8}	1.09×10^{-2}	30000	0
Cobalt-60	5.30	1.43×10^{-4}	1.71×10^{1}	3	1.33×10^{1}
Palladium-107	7.00×10^{6}	5.60×10^{-14}	6.67×10^{-9}	3000	0
Silver-108m	4.38×10^{2}	9.62×10^{-6}	1.15	30000	0
Cadmium-109	1.30	1.24×10^{-23}	1.48×10^{-18}	30	0
Cadmium-113	1.40×10^{1}	8.48×10^{-8}	1.01×10^{-2}	30000	0
Promethium-147	2.70	3.81×10^{-17}	4.53×10^{-12}	30000	0
Samarium-146	1.20×10^{8}	1.92×10^{-12}	2.29×10^{-7}	30	0
Samarium-151	9.00×10^{1}	2.75×10^{-3}	3.27×10^{2}	30000	0
Europium-152	1.20×10^{1}	2.04×10^{-2}	2.42×10^{3}	3	1.16×10^{2}
Europium-154	1.60×10^{1}	2.06×10^{-3}	2.45×10^{2}	3	1.02×10^{2}
Europium-155	1.80	7.13×10^{-21}	8.49×10^{-16}	30	0
Terbium-158	1.50×10^{2}	1.66×10^{-12}	1.97×10^{-7}	3000	0

Note that a value of zero indicates no additional decay time is needed to reach the ANSI Screening Level for that isotope

As seen in the tables, it will take over 80 years (2102) for the Cobalt-60 to decay to the screening levels. It will take thousands to millions of years for all isotopes to decay to this level. The longer-lived isotopes do not pose an external exposure hazard and likely pose little internal hazard as they are activation products in a solid matrix.

5.2 RADIATION EXPOSURE CALCULATIONS AND SCENARIO ASSUMPTIONS

Radiation exposure is considered from two perspectives in this paper. The first is for an industrial worker spending approximately 40 hours per week performing work duties on a parking lot type surface directly over the buried entombment. This is conservative as it is highly unlikely a worker would be stationary in such a small area for 40 hours per week.

- The majority of the radioactivity in the entombment is contained in the main source. Per the SAR, the largest portion of the radioactivity is in the upper and lower core support plates. As a conservative measure, all radioactivity is assumed to be in the upper support plate. This 8-inch thick plate is located 22 feet below the top of the concrete plug placed on top of the reactor.
- The secondary source is shown in the SAR as being below the 80-foot elevation mark. As such, it is be modeled as a 1-inch thick steel plate located at the 80-foot elevation. Note that the top of the reactor shield plug is at the 100-foot elevation.

- The concrete source is a hollow cylinder with 6-inch thick walls. An equivalent amount of concrete is a 2-foot radius cylinder, 8 feet long, as this was the active region of the core, and was modeled as such.
- In all cases the 8-inch thick vessel head, the 12-inch thick concrete plug, the 3-inch traffic slab, and the 24-inch crystal concrete were used as shields in the modeling.
- The exposure dose point modeled is one foot above the top of the crystal concrete.
- The radioactivity of the source terms used in the modeling were decay corrected to February 2021.
- MicroShield® dose modeling software, Version10.06, was used to calculate the exposure based on the information in Table 7:

Table 7: MicroShield Parameter Input Summary

Parameter	Description/Value
Main Source Term	Table 1, Activity as of February 2021 (Ci)
Secondary Source Term	Table 3, Activity as of February 2021 (Ci)
Concrete Source Term	Table 5, Activity as of February 2021 (Ci)
Main Source Geometry	Upper core support plate: 8-foot diameter, 8 inches thick
Secondary Source Geometry	1 inch thick, 8-foot diameter, steel plate at the 80-foot elevation
Concrete Source Geometry	Concrete cylinder with a radius of 2 feet, 8 feet long
Exposure point	1 foot above the crystal concrete, center of the reactor
Shields	Reactor vessel head – 8-inch thick steel plate Concrete Plug – 12 inches thick Traffic Slab – 3 inches thick (minimum thickness) Crystalline Concrete – 24 inches thick

MicroShield dose modeling results are shown in Attachment 1.

MicroShield Dose Modeling Results:

Main Source	3.6×10^{-8} mrad/h	2.9×10^{-5} μrad/h
Secondary Source	1.5×10^{-8} mrad/h	1.5×10^{-5} μrad/h
Concrete Source	2.1×10^{-10} mrad/h	2.1×10^{-7} μrad/h
Total	5.1×10^{-8} mrad/h	5.1×10^{-5} μrad/h

Modeling dose results shown above include buildup. None of these dose rates are detectable. From a sensitivity perspective, if the source is moved up to just under the reactor vessel head, then the exposure from the main source would increase to 1.6×10^{-7} mrad/h or 1.6×10^{-4} μrad/h. The dose modeling results and the bounding dose value are not considered to be detectable.

For the occupational exposure scenario, exposure time is 40 h/w x 50 w/y = 2,000 h/y.

Total exposure is then 5.1×10^{-5} μrad/h x 2,000 h/y = 0.1 μrad/y or 1×10^{-4} mrad/y (1×10^{-4} mrem/y).

10 CFR 835.2 defines a radiation weighting factor for converting rad to rem. In the weighting factor table, this factor is equal to 1 for photons. The only direct radiation exposure at PNPF is photons. Therefore, 1 mrad = 1 mrem.

This is orders of magnitude less than the 100 mrem/y public dose limit, and the decommissioning limit. It is 0.001% of the ANSI Screening level based on 1 mrem/y.

The bounding estimate of 1.6×10^{-4} µrad/h (1.6×10^{-4} µrem/h) will result in an annual exposure of approximately 3.2×10^{-4} mrem/y which is still orders of magnitude below the public and decommissioning limits. This bounding estimate is also less than 0.05% of the 1 mrem/y screening level used by the ANSI Standard. Note that concentration of the material directly under the reactor vessel head is not physically possible.

The second radiation exposure perspective presented in this paper is the radiation exposure to a resident living on the site directly over the buried entombment, spending approximately 16 hours per day.

For the resident scenario, exposure time is 16 h/d x 366 d/y, or 5,856 h/y.

The total resident exposure is 5.1×10^{-5} µrad/h x 5,856 h/y = 0.3 µrad/y or 3×10^{-4} mrem/y. This is orders of magnitude below the public dose limit and the decommissioning limit. It is also a fraction of a percent of the ANSI Screening Level. Note that this is a highly unlikely scenario as the area is currently in an industrial area near the City's wastewater treatment plant, and is owned by the Department of Energy with appropriate controls in place to prohibit residential use of the property. Even if a residence was allowed, the exposure from the entombment would be negligible.

5.3 INDIRECT EXPOSURE

Indirect radiation exposure is possible when radioactive material is released from the entombment to the environment. In order for this to happen, water intrusion would need to occur. This will require the breach of the concrete outer shell (8 feet thick), penetration through the sealed reactor vessel, corrosion of the steel structural members, and mobilization of radionuclides back out into the environment.

5.3.1 Breach of Entombment

The entombment consists of the reactor complex which includes the bioshield, spent fuel pool, top concrete floor plugs, waterproof coatings, and a poured concrete cover. The top plug over the reactor was welded in place as part of the decommissioning process. The entombment is expected to be coated with an additional 2 feet of long-lived, waterproofing crystalline concrete as part of the demolition project, resulting in a total concrete barrier annular thickness of 10 feet encapsulating the entombment.

5.3.1.1 Design Features

The dismantlement specification (SS745N20001) for the reactor complex lists the following:

- Radioactive materials are present within the reactor complex and will be left in place and will radioactively decay. The reactor complex will be made radiologically safe for the duration of the decay period by permanently closing it to personnel access except as might be gained through deliberate and extensive use of equipment such as jack hammers, pneumatic drills, cutting torches, and explosives.
- The release of radioactive materials from the reactor complex will be prevented by stopping water from flowing into the reactor complex, picking up radioactivity and flowing back out of the complex.

In order to accomplish this, all piping into and out of the complex was cut off and plugs were welded inside the pipes. All penetrations were filled with concrete. The reactor vessel was filled with dry sand. A gasket was installed and the reactor vessel head was bolted on, in the same manner as would be done during reactor operations. After all penetrations were sealed with concrete, the entire structure was painted with a water-resistant paint. A waterproof coating was applied to the top of the reactor complex and a slab was poured on top of that.

In the years since the reactor complex was stabilized and sealed, the entire complex has been inside the reactor building dome. This dome prevents any impacts from weather on the complex. There are no records of flooding in or around the reactor building dome since 1967.

The reactor building dome will be removed as part of the Piqua Demolition Project. The specifications for this demolition include requirements for the demolition contractor to protect the entombment from damage during demolition. As part of the demolition project, crystalline concrete will be added to the entombment to further seal any cracks or other potential penetrations. The entire area around the entombment will be backfilled and sloped to prevent pooling of water on or around the entombment.

These design features, taken in total, provide assurance that water intrusion into the reactor cavity and reactor vessel will be prevented, and any such intrusion would be considered highly improbable.

5.3.1.2 Potential Breaching Scenarios

There are several natural and manmade events requiring evaluation that have the potential to cause a failure of the entombment and for water intrusion to occur. These evaluations are presented below.

<u>Earthquake</u>

In Chapter II of the SAR, Section 2, is a description of the seismology for the Piqua area. To date, no earthquake damage has been noted at Piqua. The reactor was designed to withstand intensities of 6 to 8 on the Rossi-Forel scale (similar to the Mercalli scale). Although the SAR references Seismic Zone 2, that reference is considered dated. The current U.S. Geological Data shows the Piqua area to be Mercalli zone III to V, which is minimal damage to structures.

Based on the design and construction of the reactor complex, the additional features added to seal openings, and the lack of potential for significant earthquakes in the region, earthquake damage to the entombment is not considered to be a credible scenario.

Tornado

The SAR reports an average of 3.2 tornados per year in the State of Ohio. While this is a low frequency, it is a potential threat to the Piqua area.

Once the demolition project is complete, there will be no above ground structures associated with the entombment. This essentially takes the potential damage from a tornado to zero.

Flood

From the SAR, flood control of the Miami River was established in 1921. A retarding basin and dam were established four miles above Piqua for flood protection. This dam, together with levees and river channel improvements at Piqua, was designed to permit a maximum river flow at Piqua of 80,000 cubic feet per second (cfs). The elevation of the river at the reactor site during a flow of this magnitude will not exceed 864.5 feet above sea level. The highest river flow on record at Piqua after the construction of the dam was 22,000 cfs in 1929 and 1933. The elevation of the river at the reactor site during these flows was 857.4 feet above sea level. This is 8.6 feet below the reactor building main floor level. The arbitrary 100-foot level of the reactor building corresponds to 866 feet above sea level. This 866-foot elevation corresponds to the top of the entombment.

Given the elevation of the entombment, it is not credible for a flood to cover the area. In the event of a beyond design basis flood, the water will recede quickly (1 to 2 days) which will limit the potential for water seepage into the entombment. This short time coupled with the sealed entombment will prevent any significant water intrusion.

Flooding of the entombment or significant water intrusion into the entombment from the Miami River is not considered to be a credible scenario.

Inadvertent Drilling

There is a potential in the far future that the City of Piqua, OH, or the DOE might no longer control the land/property where the reactor entombment is buried or that current land/deed restrictions are not followed or implemented in the fashion they are today. At this point, the future land owner might attempt to drill a well to reach potable water. While this is highly unlikely to occur, as the river is adjacent to the property and would be a much easier source for water, it was not completely ruled out as a potential entombment breach scenario. After evaluation, it was concluded that an attempt to drill through the entombment structure would most likely result in drill refusal. In most all situations today (and likely into the future), once a drill refusal occurs, the well location is simply moved to a new location where the drilling does not encounter a refusal and drilling can reach the aquifer.

Breaching the entombment through inadvertent drilling is not considered credible.

Deliberate Attempts to Breach

Once the City of Piqua no longer controls the site, there is a potential for persons to make a deliberate attempt to open the entombment, despite the warnings posted. The SAR also addresses this potential where extraordinary methods such as explosives are used. It is unlikely that deliberate attempts can be prevented.

This deliberate breach is not considered credible as long as the City of Piqua exists and is in control of the area. This will give sufficient time for the Co-60 source term to decay to levels below the screening level. The remaining gamma emitters, primarily Ag-110m, have much lower dose rates. In addition, the other remaining radionuclides are all matrixed in the steel and concrete such that minimal, if any, internal exposure is possible.

Therefore, while deliberate breach of the entombment is possible, the risk of significant personnel exposure is minimal.

5.3.2 Water Intrusion

Water intrusion is not considered to be a credible event once the demolition of the reactor building and the stabilization of the entombment is complete. This is due to the addition of a two-foot thick crystalline concrete barrier over the entombment. This crystalline concrete is considered waterproof as adding water only increases the crystalline structure and makes it more impervious. This concrete is considered waterproof even for full immersion projects such as tanks. It is also self-healing in the event of cracks. The addition of this barrier, virtually eliminates the risk of water entering the entombment.

Even in the unlikely event that water intrusion to the entombment occurs, the risk of radioactive material escaping to the environment is small.

The reactor vessel and upper cavity are filled with dry sand. While this will not prevent water flow, it does provide a volume of material that will impede the movement of radioactive particulates. Sand is known to be an excellent particulate filter and would trap most particulates. As part of the PNPF retirement, an analysis of the potential for corrosion was written by Atomics International, TI-745-20-005; *PNPF Retirement, Corrosion of Steel Surfaces.* This document quotes a study that showed steel in contact with silica (sand) corrodes at a much lower rate than steel covered only by water.

The corrosion study also shows that corrosion of the steel reactor vessel and core internals is a very slow process. For example, if the steel was in a completely saturated environment, corrosion would be 0.002 to 0.006 inches per year. The total weight expected to corrode per year is estimated at 0.865 ounces per square foot (24.5 grams per square foot). The source terms listed in section 5.1 have already been converted to pCi/g which will allow a worst case determination of material at risk. However, this assumes a complete submersion of the metal in water with no silica present, which is contrary to current situation. Any leak into the reactor would likely be very slow as a catastrophic failure of the vessel is not considered a credible scenario. Not only would water intrusion be a slow process, but then diffusion and migration of material out of the entombment would also be a very slow process (even neglecting the effects of the sand material in the entombment).

The corrosion processes described above apply to the iron component of the steel. Materials such as nickel, silver, and cobalt do not corrode in water. As such, isotopes of these metals would not corrode, dissolve, and then be available for dispersion.

If a pathway for water intrusion into the reactor develops, then the corrosion process would be slow, and it would require that a large volume of water be present to drive it. In addition, the corrosion of most of the entombment material would be minimal.

Risk from dispersion of radioactive material from within the entombment to the environment is considered insignificant or negligible.

6.0 SUMMARY

The results of the dose modeling performed in this report validate that the external radiation exposure risk to personnel working or living at the former PNPF site (directly above where the entombment is buried) are significantly below the naturally-occurring background radiation levels in the Piqua area and are orders of magnitude below current regulatory dose limits established for members of the public. In addition, it has been concluded that a release to the environment of the potential radioactive material currently entombed at former PNPF site release to the environment of the radioactive material entombed at Piqua is considered extremely unlikely and poses no insignificant or negligible risk to personnel working at or living on the former PNPF site.

7.0 REFERENCES

AI-1968a. AI-AEC-MEMO-12707. Piqua Nuclear Power Facility, Dismantlement Plan. Atomics International. August 1968

AI-1968b. AI-AEC-MEMO-122708. Piqua Nuclear Power Facility Retirement, Safety Analysis Report. Atomics International. August 1968

AI-1968c. TI-745-20-001. Piqua Nuclear Power Facility Retirement, Description of Reactor Complex. Atomics International. September 1968

AI-1968d. TI-745-20-005. PNPF Retirement, Corrosion of Steel Surfaces. Atomics International. September, 1968

ATTACHMENT 1

MicroShield Results

MicroShield 10.06
NV5 Dade Moeller

Date	By	Checked

File Name	Run Date	Run Time	Duration
Main Source term - all shields.msd	June 30, 2021	8:24:24 AM	00:00:00

Project Info	
Case Title	Piqua External Main
Description	External Exp - Main Source Term in Vessel - all shields
Geometry	8 - Cylinder Volume - End Shields

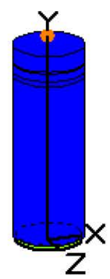
Source Dimensions	
Height	20.32 cm (8.0 in)
Radius	121.92 cm (4 ft)

Dose Points

A	X	Y	Z
#1	0.0 cm (0 in)	789.94 cm (25 ft 11.0 in)	0.0 cm (0 in)

Shields

Shield N	Dimension	Material	Density (g/cm^3)
Source	5.79e+04 inģ	Iron	7.86
Shield 1	244.0 in	Air	0.00122
Shield 2	8.0 in	Iron	7.86
Shield 3	12.0 in	Concrete	2.35
Shield 4	3.0 in	Concrete	2.35
Shield 5	24.0 in	Concrete	2.35
Air Gap		Air	0.00122



Source Input: Grouping Method - Actual Photon Energies
Library: Grove

Nuclide	Ci	Bq	ĩCi/cmģ	Bq/cmģ
Be-10	3.8100e-006	1.4097e+005	4.0151e-006	1.4856e-001
C-14	5.7200e-005	2.1164e+006	6.0280e-005	2.2304e+000
Co-60	1.0800e+000	3.9960e+010	1.1382e+000	4.2112e+004
Fe-55	4.8400e-005	1.7908e+006	5.1006e-005	1.8872e+000

Buildup: The material reference is Shield 2.
Integration Parameters

Radial	20
Circumferential	10
Y Direction (axial)	10

Results

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm^2/sec No Buildup	Fluence Rate MeV/cm^2/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.0006	7.516e+03	0.000e+00	5.462e-33	0.000e+00	2.615e-32	0.000e+00	2.283e-32	0.000e+00	2.283e-34
0.0059	1.475e+05	0.000e+00	9.861e-31	0.000e+00	5.132e-31	0.000e+00	4.480e-31	0.000e+00	4.480e-33
0.0059	2.915e+05	0.000e+00	1.953e-30	0.000e+00	1.014e-30	0.000e+00	8.854e-31	0.000e+00	8.854e-33
0.0065	5.883e+04	0.000e+00	4.335e-31	0.000e+00	2.047e-31	0.000e+00	1.787e-31	0.000e+00	1.787e-33
0.6938	6.518e+06	1.094e-14	1.111e-12	2.112e-17	2.145e-15	1.843e-17	1.873e-15	1.843e-19	1.873e-17
1.1732	3.996e+10	1.020e-07	4.653e-06	1.824e-10	8.314e-09	1.592e-10	7.258e-09	1.592e-12	7.258e-11
1.3325	3.996e+10	5.086e-07	1.907e-05	8.825e-10	3.309e-08	7.704e-10	2.889e-08	7.704e-12	2.889e-10
Total	**7.993e+10**	**6.107e-07**	**2.372e-05**	**1.065e-09**	**4.140e-08**	**9.296e-10**	**3.614e-08**	**9.296e-12**	**3.614e-10**

MicroShield 10.06
NV5 Dade Moeller

Date	By	Checked

File Name	Run Date	Run Time	Duration
Scondary Source Term - all shields.msd	June 30, 2021	8:40:25 AM	00:00:00

Project Info	
Case Title	Piqua External Sec
Description	External Exp - Secondary Source Term - all shields
Geometry	8 - Cylinder Volume - End Shields

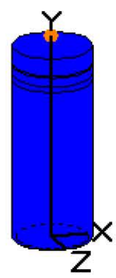
Source Dimensions	
Height	2.54 cm (1.0 in)
Radius	121.92 cm (4 ft)

Dose Points

A	X	Y	Z
#1	0.0 cm (0 in)	711.2 cm (23 ft 4.0 in)	0.0 cm (0 in)

Shields

Shield N	Dimension	Material	Density (g/cm^3)
Source	7238.23 in³	Iron	7.86
Shield 1	220.0 in	Air	0.00122
Shield 2	8.0 in	Iron	7.86
Shield 3	12.0 in	Concrete	2.35
Shield 4	3.0 in	Concrete	2.35
Shield 5	24.0 in	Concrete	2.35
Air Gap		Air	0.00122



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

Nuclide	Ci	Bq	μCi/cm³	Bq/cm³
Ag-108m	1.1000e-002	4.0700e+008	9.2738e-002	3.4313e+003
Ar-39	3.4700e-006	1.2839e+005	2.9255e-005	1.0824e+000
C-14	1.7900e-005	6.6230e+005	1.5091e-004	5.5837e+000
Ca-41	4.4000e-004	1.6280e+007	3.7095e-003	1.3725e+002
Cl-36	1.6000e-004	5.9200e+006	1.3489e-003	4.9910e+001
Co-60	3.0300e-002	1.1211e+009	2.5545e-001	9.4517e+003
Fe-55	1.8100e-004	6.6970e+006	1.5260e-003	5.6461e+001
Mo-93	9.9100e-007	3.6667e+004	8.3549e-006	3.0913e-001
Ni-59	4.9000e-002	1.8130e+009	4.1311e-001	1.5285e+004
Ni-63	5.1500e+000	1.9055e+011	4.3418e+001	1.6065e+006

Buildup: The material reference is Shield 2.
Integration Parameters

Radial	20
Circumferential	10
Y Direction (axial)	10

Results

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm^2/sec No Buildup	Fluence Rate MeV/cm^2/sec With Buildup	Exposure Rate mR/hr	Exposure Rate mR/hr	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr

				No Buildup	With Buildup		With Buildup		With Buildup
0.015	6.472e+08	0.000e+00	1.341e-26	0.000e+00	1.150e-27	0.000e+00	1.004e-27	0.000e+00	1.004e-29
0.02	2.670e+08	0.000e+00	8.692e-27	0.000e+00	3.011e-28	0.000e+00	2.629e-28	0.000e+00	2.629e-30
0.03	8.642e+01	0.000e+00	3.864e-33	0.000e+00	3.830e-35	0.000e+00	3.343e-35	0.000e+00	3.343e-37
0.08	2.884e+07	4.501e-61	8.228e-27	7.123e-64	1.302e-29	6.218e-64	1.137e-29	6.218e-66	1.137e-31
0.4	3.658e+08	6.555e-16	9.456e-14	1.277e-18	1.842e-16	1.115e-18	1.608e-16	1.115e-20	1.608e-18
0.6	3.681e+08	4.887e-13	5.756e-11	9.538e-16	1.124e-13	8.327e-16	9.808e-14	8.327e-18	9.808e-16
0.8	3.683e+08	3.370e-11	2.731e-09	6.410e-14	5.195e-12	5.596e-14	4.536e-12	5.596e-16	4.536e-14
1.0	1.121e+09	2.207e-09	1.260e-07	4.067e-12	2.323e-10	3.551e-12	2.028e-10	3.551e-14	2.028e-12
1.5	1.121e+09	3.387e-07	1.034e-05	5.699e-10	1.739e-08	4.975e-10	1.518e-08	4.975e-12	1.518e-10
Total	**4.287e+09**	**3.410e-07**	**1.047e-05**	**5.740e-10**	**1.763e-08**	**5.011e-10**	**1.539e-08**	**5.011e-12**	**1.539e-10**

MicroShield 10.06
NV5 Dade Moeller

Date	By	Checked

File Name	Run Date	Run Time	Duration
Concrete source term - all shields.msd	June 30, 2021	1:21:04 PM	00:00:00

Project Info	
Case Title	Piqua Ext Concrete
Description	External Exp - Concrete source term - all shields
Geometry	8 - Cylinder Volume - End Shields

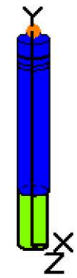
Source Dimensions	
Height	243.84 cm (8 ft)
Radius	60.96 cm (2 ft)

Dose Points

A	X	Y	Z
#1	0.0 cm (0 in)	952.5 cm (31 ft 3.0 in)	0.0 cm (0 in)

Shields

Shield N	Dimension	Material	Density (g/cm^3)
Source	1.74e+05 inģ	Concrete	2.35
Shield 1	220.0 in	Air	0.00122
Shield 2	8.0 in	Iron	7.86
Shield 3	12.0 in	Concrete	2.35
Shield 4	3.0 in	Concrete	2.35
Shield 5	24.0 in	Concrete	2.35
Air Gap		Air	0.00122



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

Nuclide	Ci	Bq	ìCi/cmģ	Bq/cmģ
Ag-108m	9.6200e-006	3.5594e+005	3.3793e-006	1.2504e-001
Al-26	1.7500e-008	6.4750e+002	6.1474e-009	2.2745e-004
Ar-39	7.9000e-002	2.9230e+009	2.7751e-002	1.0268e+003
Be-10	1.3600e-008	5.0320e+002	4.7774e-009	1.7676e-004
C-14	4.5200e-004	1.6724e+007	1.5878e-004	5.8748e+000
Ca-41	4.5500e-003	1.6835e+008	1.5983e-003	5.9138e+001
Cd-109	1.2400e-023	4.5880e-013	4.3559e-024	1.6117e-019
Cd-113	8.4800e-008	3.1376e+003	2.9789e-008	1.1022e-003
Co-60	1.4300e-004	5.2910e+006	5.0233e-005	1.8586e+000
Eu-152	2.0400e-002	7.5480e+008	7.1661e-003	2.6515e+002
Eu-154	2.0600e-003	7.6220e+007	7.2364e-004	2.6775e+001
Eu-155	7.1300e-021	2.6381e-010	2.5046e-021	9.2672e-017
Fe-55	9.1300e-008	3.3781e+003	3.2072e-008	1.1867e-003
H-3	4.5100e-001	1.6687e+010	1.5843e-001	5.8618e+003
K-40	1.8200e-006	6.7340e+004	6.3933e-007	2.3655e-002
Na-22	1.1900e-012	4.4030e-002	4.1802e-013	1.5467e-008
Pd-107	5.6000e-014	2.0720e-003	1.9672e-014	7.2786e-010
Pm-147				
Sm-151				

Buildup: The material reference is Source.

Integration Parameters	
Radial	20
Circumferential	10
Y Direction (axial)	10

Results

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm^2/sec No Buildup	Fluence Rate MeV/cm^2/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.015	1.413e+08	0.000e+00	2.424e-27	0.000e+00	2.079e-28	0.000e+00	1.815e-28	0.000e+00	1.815e-30
0.02	2.335e+05	0.000e+00	6.301e-30	0.000e+00	2.183e-31	0.000e+00	1.905e-31	0.000e+00	1.905e-33
0.03	7.558e-02	0.000e+00	4.510e-36	0.000e+00	4.470e-38	0.000e+00	3.902e-38	0.000e+00	3.902e-40
0.04	4.621e+08	5.265e-302	7.313e-26	2.329e-304	3.234e-28	2.033e-304	2.824e-28	2.033e-306	2.824e-30
0.05	1.156e+08	6.539e-169	5.981e-26	1.742e-171	1.593e-28	1.521e-171	1.391e-28	1.521e-173	1.391e-30
0.06	3.421e-12	7.304e-129	1.271e-44	1.451e-131	2.524e-47	1.267e-131	2.203e-47	1.267e-133	2.203e-49
0.08	2.522e+04	6.912e-65	2.229e-28	1.094e-67	3.527e-31	9.549e-68	3.079e-31	9.549e-70	3.079e-33
0.1	2.454e+08	3.259e-43	1.002e-23	4.986e-46	1.533e-26	4.353e-46	1.339e-26	4.353e-48	1.339e-28
0.2	6.177e+07	1.361e-24	2.651e-21	2.403e-27	4.678e-24	2.097e-27	4.084e-24	2.097e-29	4.084e-26
0.3	2.041e+08	8.110e-20	9.510e-17	1.538e-22	1.804e-19	1.343e-22	1.575e-19	1.343e-24	1.575e-21
0.4	4.830e+07	4.201e-18	2.711e-15	8.185e-21	5.282e-18	7.145e-21	4.612e-18	7.145e-23	4.612e-20
0.5	4.424e+06	1.648e-17	6.447e-15	3.234e-20	1.265e-17	2.823e-20	1.105e-17	2.823e-22	1.105e-19
0.6	3.848e+07	2.533e-15	6.482e-13	4.944e-18	1.265e-15	4.316e-18	1.104e-15	4.316e-20	1.104e-17
0.8	1.644e+08	7.715e-13	1.037e-10	1.467e-15	1.972e-13	1.281e-15	1.722e-13	1.281e-17	1.722e-15
1.0	3.517e+08	3.678e-11	3.021e-09	6.779e-14	5.569e-12	5.918e-14	4.862e-12	5.918e-16	4.862e-14
1.5	2.101e+08	3.678e-09	1.363e-07	6.189e-12	2.293e-10	5.403e-12	2.002e-10	5.403e-14	2.002e-12
2.0	6.459e+02	2.470e-13	5.766e-12	3.820e-16	8.917e-15	3.334e-16	7.785e-15	3.334e-18	7.785e-17
3.0	1.554e+00	2.087e-14	2.786e-13	2.831e-17	3.780e-16	2.471e-17	3.300e-16	2.471e-19	3.300e-18
Total	**2.048e+09**	**3.716e-09**	**1.394e-07**	**6.258e-12**	**2.351e-10**	**5.463e-12**	**2.052e-10**	**5.463e-14**	**2.052e-12**

MicroShield 10.06 NV5 Dade Moeller

Date	By	Checked

File Name	Run Date	Run Time	Duration
Source closer - all shields.msd	June 30, 2021	1:31:40 PM	00:00:00

Project Info	
Case Title	Piqua External
Description	External exp source closer - all shields
Geometry	8 - Cylinder Volume - End Shields

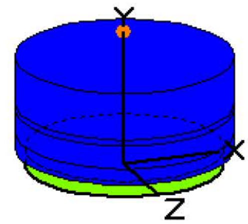
Source Dimensions	
Height	20.32 cm (8.0 in)
Radius	121.92 cm (4 ft)

Dose Points

A	X	Y	Z
#1	0.0 cm (0 in)	170.18 cm (5 ft 7.0 in)	0.0 cm (0 in)

Shields

Shield N	Dimension	Material	Density (g/cm^3)
Source	9.49e+05 cm³	Iron	7.86
Shield 1	20.32 cm	Iron	7.86
Shield 2	30.48 cm	Concrete	2.35
Shield 3	7.62 cm	Concrete	2.35
Shield 4	60.96 cm	Concrete	2.35
Air Gap		Air	0.00122



Source Input: Grouping Method - Actual Photon Energies
Library: Grove

Nuclide	Ci	Bq	µCi/cm³	Bq/cm³
Ag-108m	1.1000e-002	4.0700e+008	1.1592e-002	4.2891e+002
Co-60	1.0800e+000	3.9960e+010	1.1382e+000	4.2112e+004

Buildup: The material reference is Shield 1.
Integration Parameters

Radial	20
Circumferential	10
Y Direction (axial)	10

Results

Energy (MeV)	Activity (Photons/sec)	Fluence Rate MeV/cm^2/sec No Buildup	Fluence Rate MeV/cm^2/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mR/hr With Buildup	Absorbed Dose Rate mrad/hr No Buildup	Absorbed Dose Rate mrad/hr With Buildup	Absorbed Dose Rate mGy/hr No Buildup	Absorbed Dose Rate mGy/hr With Buildup
0.0028	1.959e+07	0.000e+00	1.203e-27	0.000e+00	1.298e-27	0.000e+00	1.133e-27	0.000e+00	1.133e-29
0.003	2.338e+06	0.000e+00	1.506e-28	0.000e+00	1.549e-28	0.000e+00	1.352e-28	0.000e+00	1.352e-30
0.021	7.417e+07	0.000e+00	3.974e-26	0.000e+00	1.177e-27	0.000e+00	1.028e-27	0.000e+00	1.028e-29
0.0212	1.407e+08	0.000e+00	7.591e-26	0.000e+00	2.197e-27	0.000e+00	1.918e-27	0.000e+00	1.918e-29
0.022	2.091e+06	0.000e+00	1.163e-27	0.000e+00	2.991e-29	0.000e+00	2.611e-29	0.000e+00	2.611e-31
0.0222	3.961e+06	0.000e+00	2.216e-27	0.000e+00	5.560e-29	0.000e+00	4.854e-29	0.000e+00	4.854e-31
0.0238	4.470e+07	0.000e+00	2.625e-26	0.000e+00	5.273e-28	0.000e+00	4.603e-28	0.000e+00	4.603e-30
0.0249	1.277e+06	0.000e+00	7.713e-28	0.000e+00	1.347e-29	0.000e+00	1.176e-29	0.000e+00	1.176e-31
0.0304	8.642e+01	0.000e+00	6.135e-32	0.000e+00	5.857e-34	0.000e+00	5.113e-34	0.000e+00	5.113e-36
0.0792	2.884e+07	5.721e-63	1.252e-25	9.089e-66	1.990e-28	7.934e-66	1.737e-28	7.934e-68	1.737e-30
0.4339	3.658e+08	1.007e-15	1.534e-13	1.971e-18	3.003e-16	1.721e-18	2.621e-16	1.721e-20	2.621e-18

0.6144	3.679e+08	3.119e-13	3.797e-11	6.080e-16	7.400e-14	5.308e-16	6.460e-14	5.308e-18	6.460e-16
0.6938	6.518e+06	3.623e-14	3.824e-12	6.994e-17	7.382e-15	6.106e-17	6.445e-15	6.106e-19	6.445e-17
0.7229	3.683e+08	3.819e-12	3.817e-10	7.344e-15	7.341e-13	6.412e-15	6.408e-13	6.412e-17	6.408e-15
1.1732	3.996e+10	4.068e-07	1.937e-05	7.269e-10	3.461e-08	6.346e-10	3.021e-08	6.346e-12	3.021e-10
1.3325	3.996e+10	2.127e-06	8.339e-05	3.691e-09	1.447e-07	3.222e-09	1.263e-07	3.222e-11	1.263e-09
Total	**8.135e+10**	**2.534e-06**	**1.028e-04**	**4.418e-09**	**1.793e-07**	**3.857e-09**	**1.565e-07**	**3.857e-11**	**1.565e-09**
	Sensitivity	**Variable**	**Y Dose Point 1**	**(1 of 5)**	**(1.73e+02 cm)**				
0.0028	1.959e+07	0.000e+00	1.173e-27	0.000e+00	1.265e-27	0.000e+00	1.105e-27	0.000e+00	1.105e-29
0.003	2.338e+06	0.000e+00	1.469e-28	0.000e+00	1.510e-28	0.000e+00	1.319e-28	0.000e+00	1.319e-30
0.021	7.417e+07	0.000e+00	3.875e-26	0.000e+00	1.148e-27	0.000e+00	1.002e-27	0.000e+00	1.002e-29
0.0212	1.407e+08	0.000e+00	7.402e-26	0.000e+00	2.142e-27	0.000e+00	1.870e-27	0.000e+00	1.870e-29
0.022	2.091e+06	0.000e+00	1.134e-27	0.000e+00	2.917e-29	0.000e+00	2.546e-29	0.000e+00	2.546e-31
0.0222	3.961e+06	0.000e+00	2.161e-27	0.000e+00	5.422e-29	0.000e+00	4.734e-29	0.000e+00	4.734e-31
0.0238	4.470e+07	0.000e+00	2.560e-26	0.000e+00	5.142e-28	0.000e+00	4.489e-28	0.000e+00	4.489e-30
0.0249	1.277e+06	0.000e+00	7.522e-28	0.000e+00	1.313e-29	0.000e+00	1.147e-29	0.000e+00	1.147e-31
0.0304	8.642e+01	0.000e+00	5.982e-32	0.000e+00	5.711e-34	0.000e+00	4.986e-34	0.000e+00	4.986e-36
0.0792	2.884e+07	5.718e-63	1.221e-25	9.084e-66	1.940e-28	7.930e-66	1.694e-28	7.930e-68	1.694e-30
0.4339	3.658e+08	1.007e-15	1.534e-13	1.971e-18	3.002e-16	1.720e-18	2.621e-16	1.720e-20	2.621e-18
0.6144	3.679e+08	3.118e-13	3.795e-11	6.078e-16	7.398e-14	5.306e-16	6.458e-14	5.306e-18	6.458e-16
0.6938	6.518e+06	3.622e-14	3.822e-12	6.992e-17	7.380e-15	6.104e-17	6.443e-15	6.104e-19	6.443e-17
0.7229	3.683e+08	3.818e-12	3.816e-10	7.342e-15	7.339e-13	6.410e-15	6.407e-13	6.410e-17	6.407e-15
1.1732	3.996e+10	4.066e-07	1.936e-05	7.266e-10	3.459e-08	6.343e-10	3.020e-08	6.343e-12	3.020e-10
1.3325	3.996e+10	2.126e-06	8.334e-05	3.689e-09	1.446e-07	3.220e-09	1.262e-07	3.220e-11	1.262e-09
Total	**8.135e+10**	**2.533e-06**	**1.027e-04**	**4.415e-09**	**1.792e-07**	**3.855e-09**	**1.564e-07**	**3.855e-11**	**1.564e-09**
	Sensitivity	**Variable**	**Y Dose Point 1**	**(2 of 5)**	**(3.25e+02 cm)**				
0.0028	1.959e+07	0.000e+00	3.704e-28	0.000e+00	3.996e-28	0.000e+00	3.488e-28	0.000e+00	3.488e-30
0.003	2.338e+06	0.000e+00	4.638e-29	0.000e+00	4.769e-29	0.000e+00	4.163e-29	0.000e+00	4.163e-31
0.021	7.417e+07	0.000e+00	1.224e-26	0.000e+00	3.625e-28	0.000e+00	3.164e-28	0.000e+00	3.164e-30
0.0212	1.407e+08	0.000e+00	2.337e-26	0.000e+00	6.764e-28	0.000e+00	5.905e-28	0.000e+00	5.905e-30
0.022	2.091e+06	0.000e+00	3.582e-28	0.000e+00	9.210e-30	0.000e+00	8.040e-30	0.000e+00	8.040e-32
0.0222	3.961e+06	0.000e+00	6.824e-28	0.000e+00	1.712e-29	0.000e+00	1.495e-29	0.000e+00	1.495e-31
0.0238	4.470e+07	0.000e+00	8.082e-27	0.000e+00	1.624e-28	0.000e+00	1.417e-28	0.000e+00	1.417e-30
0.0249	1.277e+06	0.000e+00	2.375e-28	0.000e+00	4.148e-30	0.000e+00	3.621e-30	0.000e+00	3.621e-32
0.0304	8.642e+01	0.000e+00	1.889e-32	0.000e+00	1.803e-34	0.000e+00	1.574e-34	0.000e+00	1.574e-36
0.0792	2.884e+07	5.547e-63	3.856e-26	8.813e-66	6.127e-29	7.694e-66	5.348e-29	7.694e-68	5.348e-31
0.4339	3.658e+08	9.344e-16	1.414e-13	1.828e-18	2.766e-16	1.596e-18	2.415e-16	1.596e-20	2.415e-18
0.6144	3.679e+08	2.818e-13	3.394e-11	5.492e-16	6.616e-14	4.794e-16	5.776e-14	4.794e-18	5.776e-16
0.6938	6.518e+06	3.235e-14	3.374e-12	6.245e-17	6.515e-15	5.452e-17	5.687e-15	5.452e-19	5.687e-17
0.7229	3.683e+08	3.396e-12	3.353e-10	6.530e-15	6.448e-13	5.701e-15	5.629e-13	5.701e-17	5.629e-15
1.1732	3.996e+10	3.397e-07	1.589e-05	6.071e-10	2.839e-08	5.300e-10	2.478e-08	5.300e-12	2.478e-10
1.3325	3.996e+10	1.741e-06	6.692e-05	3.020e-09	1.161e-07	2.636e-09	1.014e-07	2.636e-11	1.014e-09
Total	**8.135e+10**	**2.080e-06**	**8.280e-05**	**3.627e-09**	**1.445e-07**	**3.166e-09**	**1.261e-07**	**3.166e-11**	**1.261e-09**
	Sensitivity	**Variable**	**Y Dose Point 1**	**(3 of 5)**	**(4.78e+02 cm)**				
0.0028	1.959e+07	0.000e+00	1.727e-28	0.000e+00	1.863e-28	0.000e+00	1.626e-28	0.000e+00	1.626e-30
0.003	2.338e+06	0.000e+00	2.162e-29	0.000e+00	2.223e-29	0.000e+00	1.941e-29	0.000e+00	1.941e-31
0.021	7.417e+07	0.000e+00	5.705e-27	0.000e+00	1.690e-28	0.000e+00	1.475e-28	0.000e+00	1.475e-30
0.0212	1.407e+08	0.000e+00	1.090e-26	0.000e+00	3.153e-28	0.000e+00	2.753e-28	0.000e+00	2.753e-30
0.022	2.091e+06	0.000e+00	1.670e-28	0.000e+00	4.294e-30	0.000e+00	3.748e-30	0.000e+00	3.748e-32
0.0222	3.961e+06	0.000e+00	3.181e-28	0.000e+00	7.982e-30	0.000e+00	6.968e-30	0.000e+00	6.968e-32
0.0238	4.470e+07	0.000e+00	3.768e-27	0.000e+00	7.569e-29	0.000e+00	6.608e-29	0.000e+00	6.608e-31
0.0249	1.277e+06	0.000e+00	1.107e-28	0.000e+00	1.934e-30	0.000e+00	1.688e-30	0.000e+00	1.688e-32

0.0304	8.642e+01	0.000e+00	8.807e-33	0.000e+00	8.407e-35	0.000e+00	7.339e-35	0.000e+00	7.339e-37
0.0792	2.884e+07	5.340e-63	1.798e-26	8.484e-66	2.856e-29	7.406e-66	2.493e-29	7.406e-68	2.493e-31
0.4339	3.658e+08	7.111e-16	1.062e-13	1.392e-18	2.078e-16	1.215e-18	1.814e-16	1.215e-20	1.814e-18
0.6144	3.679e+08	2.054e-13	2.439e-11	4.003e-16	4.753e-14	3.495e-16	4.150e-14	3.495e-18	4.150e-16
0.6938	6.518e+06	2.321e-14	2.386e-12	4.482e-17	4.607e-15	3.913e-17	4.022e-15	3.913e-19	4.022e-17
0.7229	3.683e+08	2.424e-12	2.358e-10	4.662e-15	4.535e-13	4.070e-15	3.959e-13	4.070e-17	3.959e-15
1.1732	3.996e+10	2.277e-07	1.049e-05	4.068e-10	1.874e-08	3.552e-10	1.636e-08	3.552e-12	1.636e-10
1.3325	3.996e+10	1.147e-06	4.345e-05	1.991e-09	7.539e-08	1.738e-09	6.581e-08	1.738e-11	6.581e-10
Total	**8.135e+10**	**1.375e-06**	**5.394e-05**	**2.397e-09**	**9.413e-08**	**2.093e-09**	**8.217e-08**	**2.093e-11**	**8.217e-10**
	Sensitivity	**Variable**	**Y Dose Point 1**	**(4 of 5)**	**(6.30e+02 cm)**				
0.0028	1.959e+07	0.000e+00	9.926e-29	0.000e+00	1.071e-28	0.000e+00	9.348e-29	0.000e+00	9.348e-31
0.003	2.338e+06	0.000e+00	1.243e-29	0.000e+00	1.278e-29	0.000e+00	1.116e-29	0.000e+00	1.116e-31
0.021	7.417e+07	0.000e+00	3.279e-27	0.000e+00	9.714e-29	0.000e+00	8.481e-29	0.000e+00	8.481e-31
0.0212	1.407e+08	0.000e+00	6.264e-27	0.000e+00	1.813e-28	0.000e+00	1.583e-28	0.000e+00	1.583e-30
0.022	2.091e+06	0.000e+00	9.599e-29	0.000e+00	2.468e-30	0.000e+00	2.155e-30	0.000e+00	2.155e-32
0.0222	3.961e+06	0.000e+00	1.829e-28	0.000e+00	4.588e-30	0.000e+00	4.006e-30	0.000e+00	4.006e-32
0.0238	4.470e+07	0.000e+00	2.166e-27	0.000e+00	4.351e-29	0.000e+00	3.799e-29	0.000e+00	3.799e-31
0.0249	1.277e+06	0.000e+00	6.365e-29	0.000e+00	1.112e-30	0.000e+00	9.704e-31	0.000e+00	9.704e-33
0.0304	8.642e+01	0.000e+00	5.063e-33	0.000e+00	4.833e-35	0.000e+00	4.219e-35	0.000e+00	4.219e-37
0.0792	2.884e+07	4.889e-63	1.034e-26	7.767e-66	1.642e-29	6.781e-66	1.433e-29	6.781e-68	1.433e-31
0.4339	3.658e+08	5.022e-16	7.444e-14	9.828e-19	1.457e-16	8.580e-19	1.272e-16	8.580e-21	1.272e-18
0.6144	3.679e+08	1.416e-13	1.669e-11	2.760e-16	3.253e-14	2.410e-16	2.840e-14	2.410e-18	2.840e-16
0.6938	6.518e+06	1.588e-14	1.620e-12	3.066e-17	3.127e-15	2.676e-17	2.730e-15	2.676e-19	2.730e-17
0.7229	3.683e+08	1.654e-12	1.596e-10	3.180e-15	3.070e-13	2.776e-15	2.680e-13	2.776e-17	2.680e-15
1.1732	3.996e+10	1.506e-07	6.891e-06	2.692e-10	1.231e-08	2.350e-10	1.075e-08	2.350e-12	1.075e-10
1.3325	3.996e+10	7.535e-07	2.834e-05	1.307e-09	4.918e-08	1.141e-09	4.293e-08	1.141e-11	4.293e-10
Total	**8.135e+10**	**9.042e-07**	**3.524e-05**	**1.576e-09**	**6.149e-08**	**1.376e-09**	**5.368e-08**	**1.376e-11**	**5.368e-10**
	Sensitivity	**Variable**	**Y Dose Point 1**	**(5 of 5)**	**(7.82e+02 cm)**				
0.0028	1.959e+07	0.000e+00	6.440e-29	0.000e+00	6.948e-29	0.000e+00	6.066e-29	0.000e+00	6.066e-31
0.003	2.338e+06	0.000e+00	8.066e-30	0.000e+00	8.293e-30	0.000e+00	7.240e-30	0.000e+00	7.240e-32
0.021	7.417e+07	0.000e+00	2.128e-27	0.000e+00	6.303e-29	0.000e+00	5.503e-29	0.000e+00	5.503e-31
0.0212	1.407e+08	0.000e+00	4.065e-27	0.000e+00	1.176e-28	0.000e+00	1.027e-28	0.000e+00	1.027e-30
0.022	2.091e+06	0.000e+00	6.229e-29	0.000e+00	1.602e-30	0.000e+00	1.398e-30	0.000e+00	1.398e-32
0.0222	3.961e+06	0.000e+00	1.187e-28	0.000e+00	2.977e-30	0.000e+00	2.599e-30	0.000e+00	2.599e-32
0.0238	4.470e+07	0.000e+00	1.405e-27	0.000e+00	2.823e-29	0.000e+00	2.465e-29	0.000e+00	2.465e-31
0.0249	1.277e+06	0.000e+00	4.130e-29	0.000e+00	7.212e-31	0.000e+00	6.296e-31	0.000e+00	6.296e-33
0.0304	8.642e+01	0.000e+00	3.285e-33	0.000e+00	3.136e-35	0.000e+00	2.738e-35	0.000e+00	2.738e-37
0.0792	2.884e+07	4.203e-63	6.706e-27	6.676e-66	1.065e-29	5.828e-66	9.301e-30	5.828e-68	9.301e-32
0.4339	3.658e+08	3.575e-16	5.279e-14	6.996e-19	1.033e-16	6.108e-19	9.018e-17	6.108e-21	9.018e-19
0.6144	3.679e+08	9.960e-14	1.169e-11	1.941e-16	2.279e-14	1.695e-16	1.990e-14	1.695e-18	1.990e-16
0.6938	6.518e+06	1.112e-14	1.130e-12	2.147e-17	2.182e-15	1.875e-17	1.905e-15	1.875e-19	1.905e-17
0.7229	3.683e+08	1.157e-12	1.112e-10	2.225e-15	2.139e-13	1.942e-15	1.868e-13	1.942e-17	1.868e-15
1.1732	3.996e+10	1.038e-07	4.735e-06	1.856e-10	8.461e-09	1.620e-10	7.387e-09	1.620e-12	7.387e-11
1.3325	3.996e+10	5.176e-07	1.941e-05	8.980e-10	3.367e-08	7.840e-10	2.940e-08	7.840e-12	2.940e-10
Total	**8.135e+10**	**6.215e-07**	**2.414e-05**	**1.084e-09**	**4.214e-08**	**9.460e-10**	**3.678e-08**	**9.460e-12**	**3.678e-10**

Sensitivity Analysis Summary - Y Dose Point 1

Dose Point No.	**Sensitivity**	**Sensitivity Dimension**	**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**	**Absorbed Dose Rate mrad/hr No Buildup**	**Absorbed Dose Rate mrad/hr With Buildup**	**Absorbed Dose Rate mGy/hr No Buildup**	**Absorbed Dose Rate mGy/hr With Buildup**
1	(1 of 5)	(1.73e+02 cm)	2.533e-06	1.027e-04	4.415e-09	1.792e-07	3.855e-09	1.564e-07	3.855e-11	1.564e-09

1	(2 of 5)	(3.25e+02 cm)	2.080e-06	8.280e-05	3.627e-09	1.445e-07	3.166e-09	1.261e-07	3.166e-11	1.261e-09
1	(3 of 5)	(4.78e+02 cm)	1.375e-06	5.394e-05	2.397e-09	9.413e-08	2.093e-09	8.217e-08	2.093e-11	8.217e-10
1	(4 of 5)	(6.30e+02 cm)	9.042e-07	3.524e-05	1.576e-09	6.149e-08	1.376e-09	5.368e-08	1.376e-11	5.368e-10
1	(5 of 5)	(7.82e+02 cm)	6.215e-07	2.414e-05	1.084e-09	4.214e-08	9.460e-10	3.678e-08	9.460e-12	3.678e-10

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