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Plutonium Recycle Test Reactor Characterization Activities and Results

Bruce C. Cornwell
Babcox and Wilcox Hanford Company

Nolan S. Hale
Stone and Webster Engineering Corporation

Richland, WA 99352
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Abstract:

Report contains results of PRTR core and associated structures characterization performed in January and February of 1997. Radiation survey data are presented, along with recommendations for stabilization activities before transitioning to a decontamination and decommissioning function. Recommendations are also made about handling the waste generated by the stabilization activities, and actions suggested by the Decontamination and Decommissioning organization.

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Authors

Bruce C. Cornwell
Babcox and Wilcox Hanford Company

Nolan S. Hale

Stone and Webster Engineering Corporation

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May 1997

EXECUTIVE SUMMARY

This report documents characterization activities in the Plutonium Recycle Test Reactor (PRTR) core and associated structures. Associated structures include reactor shielding, the moderator dump tank and the room it sits in, and the lower reactor space. These are the last remaining structures and equipment that directly supported reactor core operation.

Review of characterization data shows the highest dose rate inside the core to be 44 R/hr, with very low contamination levels. Gamma energy spectral analysis reveals the dose is primarily from ^{60}Co and ^{137}Cs . Contamination isotopes are ^{60}Co , ^{137}Cs , and extremely small amounts of ^{241}Am . Dose rates outside the core are negligible in normally accessible areas in the reactor dome. In another 30 yr, ^{60}Co (half-life of 5.271 yr) will have gone through 5 half-lives and be 3/100ths of its present amount. ^{137}Cs (half-life 30.17 yr) will have gone through one half-life and be at half its present concentration. ^{241}Am (half-life 432.2 yr) will have only gone through 7/100ths of a half-life and will essentially be unchanged. No nonstructural lead and very little accessible asbestos were found.

This report recommends that the lower reactor space be stabilized by cleaning it out and that all other structures evaluated in this report be left alone. Cleaning will probably be easiest and most effective using a high-efficiency particulate air (HEPA) filter vacuum cleaner. The dome structure should be maintained for another 30 yr. In that time, natural radioactive decay will substantially reduce exposure and contamination risks to workers.

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PRTR Characterization - Activities and Results

1.0 BACKGROUND

The background of the Plutonium Recycle Test Reactor (PRTR) is presented in a number of places. Appendix A is a background summary and directory to primary sources. It was written as a result of background research supporting the work reported here.

2.0 SCOPE AND OBJECTIVES

The scope of the PRTR characterization project was to gather radiological and chemical data from the core and remaining associated structures. This information will aid the decontamination and decommissioning organization in responsibly and safely protecting and dismantling this part of the reactor. Figure 1.0 is a cutaway drawing that shows relative locations for the work reported here. The goal was to gain as much information about the PRTR core and the remaining associated structures, with the minimum possible radiation exposure and cost. Specific project objectives were to:

- Determine contamination levels in dry gas space, reactor top face (top of the top shield), primary coolant piping internals, lower access space, moderator dump tank, and the space underneath the moderator dump tank
- Determine the dose rates directly in the reactor core
- Attempt to answer the following questions:
 - Is there water in the top shield?
 - How much contamination (fuel and fission products) condensed on the bottom of the reactor after the 1965 accident?
 - How much asbestos is there in the reactor?
 - Is there any heavy-water left in the reactor?
 - What is the condition of the moderator dump tank dunnage?
- Locate and quantify asbestos and nonstructural lead in the reactor core and associated structures.

This report also contains an inventory of the lead in the PRTR facility (Appendix B). Most of the lead in inventory is included in the building structure as shielding and is not removable.

Figure 1.0. PRTR Reactor Cavity Layout.

Not to scale, sizes are approximated

PRTR Reactor Cavity Layout

3.0 CHARACTERIZATION METHODS AND ACTIVITIES SUMMARY

To accomplish project objectives, four entries were made into the reactor and associated structures. Pictures of these entries are in Appendix C. The four entries were: 1) underneath the moderator dump tank, 2) into the lower reactor space, 3) into the reactor process tubes through the top face, and 4) up to the reactor core through one of the side shield plugs. Table 1.0 shows what information each entry was expected to provide (X), and what information each entry actually yielded (Y). The only objectives not met were those requiring entry into the dry gas space or access to the bottom of the reactor. For reasons stated below, access to the dry gas space and the bottom of the reactor was denied.

Inability to access these areas imposed the consequence of not being able to get some information directly. These consequences were mitigated by obtaining needed information as indicated below.

Table 1.0. Entries into the PRTR Reactor and Associated Structures.

Objective	Top entry	Side shield plug	Lower space	Below dump tank
I. Videotape and photograph all work, showing equipment conditions.	X, Y	X, Y	X, Y	X, Y
II. Determine contamination levels in:				
A. Heavy water dump tank			X, Y	
B. Primary piping internals/externals	X, Y		X	
C. Dry gas space		X	X	
D. Reactor top face (top of top shield)	X, Y			
E. Reactor bottom face (bottom of bottom shield)			X	
F. Lower access space			X, Y	
G. Space under dump tank				X, Y
III. Determine dose rates from the following:				
A. Core cumulative dose	X, Y	X, Y		
B. Through shield-plug opening, isotopic reading		X, Y		
IV. Answer (or get adequate information on) the following questions:				
A. Is there water in the top/bottom shields?	X, Y		X, Y	
B. What is the condition of the inside of the bottom gas seal? How much contamination condensed in that location during the 1965 accident?		X	X	

Table 1.0. Entries into the PRTR Reactor and Associated Structures.

Objective	Top entry	Side shield plug	Lower space	Below dump tank
C. Is there any heavy-water left in the reactor system?			X, Y	X, Y
D. What is the condition of the dump tank dunnage?				X, Y
V. Identify hazardous materials locations				
A. Lead	X, Y		X, Y	
B. Asbestos	X, Y		X, Y	

3.1 FIRST CONDITION; ITS CONSEQUENCE, AND MITIGATION

3.1.1 Condition

The thermal barrier in the lower reactor space was locked in the "UP" position. Its lifting jack, while in place, has been cut from any hydraulic system and is not operational as presently configured.

3.1.2 Consequence

Could not access the bottom of the reactor bottom shield and could not directly evaluate its contamination levels, the gas seals, or any piping internals.

3.1.3 Mitigation

Contamination on the bottom of the reactor can reasonably be inferred from contamination levels in the lower reactor space. The primary piping internal surfaces are assumed to be fairly uniform in contamination levels, therefore, this information was gained from the top entry. Investigation into plutonium levels inside the dry gas space was deferred to the side entry using gamma energy analysis and neutron detection. Gas seal condition is not crucial since the reactor will never operate again. Therefore, no evaluation was made regarding its condition. The raised thermal barrier required consideration of the reactor bottom face as separate from the lower reactor space.

3.2 SECOND CONDITION; ITS CONSEQUENCE, AND MITIGATION

3.2.1 Condition

Dose rates in the core were too high for gamma energy analysis inside the core during the top entry. Adequate shielding around the detector would have prevented contamination from getting into the process tube which is 8.2 cm (3.24 in.) inside diameter.

3.2.2 Consequence

Could not get good isotopic information from inside of the core by top entry.

3.2.3 Mitigation

Got isotopic information from the "shine" at the top of two of the reactor nozzles. Also, got isotopic information from the side of the reactor, through the shield plug thimble. Gathering isotopic data was not originally planned for the top entry, but an attempt was made to glean as much information as possible from this entry.

3.3 THIRD CONDITION; ITS CONSEQUENCE, AND MITIGATION

3.3.1 Condition

Because of concerns about possible plutonium contamination inside the dry gas space, it was decided to forego removal of the thimble until after evaluation of data noted in the report. But, following the evaluation of data, there is no guarantee that removal of the thimble will occur. Future evaluation of this report will determine if removal of the thimble is necessary.

3.3.2 Consequence

This decision, combined with the challenge of the thermal barrier being up, made it impossible to take smears of the dry gas space.

3.3.3 Mitigation

Information regarding maximum plutonium contamination in the dry gas space was gained from the side entry. Maximum plutonium contamination levels were inferred from gamma spectral data combined with neutron data. With these two measurements, it is possible, knowing thimble construction materials, to calculate the maximum amount of plutonium that could be in the vicinity of the probe. Neutron count was below background levels, indicating extremely low levels of fissile material. In retrospect, entry into the dry gas space would probably not have been

as difficult as feared. However, the administrative requirements to enter a space without firm data, and with its history of being involved in a nuclear accident, would have delayed this project considerably.

4.0 CHARACTERIZATION SUMMARIES BY ENTRY

4.1 REACTOR TOP FACE

The goals for this entry are noted in column 2, "Top Entry," of Table 1.0. The reactor's top face was accessed by lifting off the 6.8 ton inner rotating shield noted on Figure 1.0. This completely exposed 45 of the 85 process tube nozzles and partially exposed another 7. Figure 2.0 shows the locations of the process tubes entered and the area exposed by removing the inner rotating shield. The objectives were driven by both PRTR core characterization and personnel protection requirements. The objectives for this entry were to map the dose profile of the core by both elevation and relationship to the centerline of the core, evaluate the presence and condition of lead and asbestos, and determine if the water was removed from the top shield.

Dose rate profiles were made on the center tube, #1550, a mid-tube, #1354, and a fringe tube #1144. Selection of these tubes allows development of a dose profile perpendicular to the core centerline. Specific dose profile data are reported later in this document, but dose rates inside the core and shields ranged from a high of 44 R/hr just below the top shield bottom plate to 3.7 mR/hr at the top of the nozzles (with the nozzle caps removed). Field instrument readings of technical smears (100 cm²) from the top of the reactor ranged from 500 to 1000 dpm β/γ and were all <20 dpm α . Dose rates of places accessible to people were a maximum of 2 mR/hr on contact (for both open and closed windows), indicating no significant β dose. This dose rate was a contact reading on the tops of the nozzles, with the nozzle caps in place. This would give only an extremity dose in normal circumstances. Whole body dose readings were much less.

Removal of the inner rotating shield provided excellent access to process tube nozzles, but failed to expose the outlet ring header or the jumpers after they leave the nozzle area. Removal of the inner rotating shield also failed to expose the top shield cooling water connections, although one was visible. The top of the reactor is in good condition, not excessively corroded, and seems to be stable. No loose lead was found on the reactor top, although it was impossible to see the entire top shield. It was possible to see part of the outlet ring header and it appears lagged with what is probably asbestos. Each of the outlet jumpers has an asbestos and aluminum foil blanket around its resistance temperature device (RTD) location, but these are not accessible unless the outer rotating shield and perhaps the stationary ring shields are removed. To access these asbestos blankets, (1) it would be necessary to remove the outer rotating shield, (2) removal of the 31 ton outer rotating shield requires the added controls of making it a critical lift, since the polar crane is rated at 30 tons, (3) for these reasons, the amount of asbestos in this part of the reactor could not be estimated, and (4) the asbestos is stable and protected, as it should be in its present location until decontamination and decommissioning of the building begins.

Figure 2.0. PRTR Process Tube Locations.

Testing for the presence of water in the top shield was done by using a neutron activation method (see Table 1.0, IV.A). A plutonium/beryllium (PuBe) neutron source was placed in a pure copper tube. The entire probe was then lowered down a process tube (#1049) to desired elevations in the shield and left for 24 hr for each of three tests. In the presence of water, the fast neutrons emitted from the PuBe source are "thermalized" and reflected back to the source. The fast neutrons do not activate the copper tube as they leave the probe, but reflected thermal neutrons will. By counting activity of the copper tube it is possible to determine if the tube was in a neutron thermalizing environment, such as water. The probe was placed near the top of the top shield, near the bottom of the top shield, and mid-plane of the bottom shield as a blank since the bottom shield is known to be empty. The bottom shield is known to be empty because it was filled from the bottom, through the bottom plate of the tank, and its fill line has been cut. The conclusion from these tests (Appendix E) is that the top shield is empty of water.

Special tools to remove the nozzle caps were recovered from the lower reactor space. These tools were stored on top of the reactor before it was closed.

4.2 SIDE SHIELD PLUG ENTRY, -21 FT C-CELL

Goals for this entry are noted in Table 1.0, column 3 "Step Plug." Removal of a horizontal step plug from the biological shield allows dosimetry instruments to be placed next to the reactor vessel wall. The plug selected for removal was the lower of two step plugs in the -21 ft C-cell, 1L (Figure 1.0), mostly for ease of removal and because of the two fission chambers in the thimble. Data objectives for this entry were to characterize dose rates through the biological shield, develop data that correlated to top entry information, get contamination data from a surface analogous to the reflector surface, and to determine the maximum plutonium contamination in the dry gas space. The fission chambers were removed and disposed of as an accelerated cleanup action.

Specific dose rate profile data are presented later in this report. Contamination on the plug was very loose, with most of it being removed by wide area characterization smears. Field results on the technical smear (100 cm²) taken in the thimble were 60,000 dpm β/γ and <20 dpm α . Dose rates were taken at 6 in. intervals up to the vessel wall. A gamma spectrum instrument and neutron counter were also inserted to validate isotopic information taken from the top entry and to determine the maximum amount of transuranic isotopes possible in the calandria.

The biological shield step plug is bolted into a long stepped pipe, called a thimble. It extends from the outside of the biological shield, through the reflector, and up to the reactor vessel wall. The outside of the thimble is part of the dry gas space (a gasket on the outside of the biological shield made it gas tight) and it was water cooled on the inside (the space between the step plug and the inside of the thimble). Cooling water was provided by a side-stream from the reflector system, and was heavy water (D₂O). Contamination of the thimble inside surface should be similar to the reflector. There was still a small amount of water in the thimble when it was

first opened (~ 10 to 20 ml). The water itself is not radioactive and didn't appear to be contaminated, which means that the radioactive contaminants are not particularly soluble in water.

4.3 LOWER REACTOR SPACE

Goals for this entry are noted in Table 1.0, column 4 "Lower Space." This is the area where most stabilization activity is expected to take place. Data objectives were both characterization and personnel protection, to be used in developing a stabilization strategy.

Field readings for technical smears (100 cm²) range from 1,000 to 15,000 dpm β/γ and were all <20 dpm α . Contact Cutie Pie (CP) dose rates for specific equipment and areas in the space ranged from 2 mR/hr to 10 mR/hr with the window open. General field readings were <0.1 mR/hr. See Appendix F for specific radiological survey data (smears reported on the survey report are wide area smears).

At the time of this work, the lower reactor space had been barricaded off for nearly 30 yr and was quite dirty. The floor of the eastern part of the room is cluttered with a variety of tools, many of which are special tools made for working on the top face. There is no non-structural lead in the space, although three of the dump valves are shrouded with metal sheeting. This metal, while apparently not entirely lead, appears to have some lead in it. There is a small amount of loose asbestos lagging in the area immediately inside both temporary barriers. The asbestos is from a lagging removal job done long ago. The asbestos should be considered friable. It is also likely that the inlet ring header is asbestos-lagged, although, it was not visible because of the thermal barriers being locked in the "up" position.

This part of the work included opening up the moderator dump tank for inspection, to ensure all heavy water had been removed (Table 1.0, IV.C) and to establish the condition of the tank. The tank was found to be dry, and has a light layer of orange deposit on the bottom. The only smear available was a technical smear off the bottom of the tank. This represents probably the worst contamination case for the entire moderator system.

4.4 UNDERNEATH THE MODERATOR DUMP TANK

Goals for this entry are referred to on Table 1.0, column 5 "Below Dump Tank." Because of its limited access, the data objective for this part of the reactor structure was primarily contamination information to determine curie content of the room. This data objective was chosen because of very restricted access to the space, limiting the value of additional personnel protection data.

Field readings of technical smears (100 cm²) range from 2,000 to 40,000 dpm β/γ and <20 to 420 dpm α . Dose rates were about 10 mR/hr underneath the tank (instrument was RO-7, using a low range probe) and 0 mR/hr at the access tunnel opening (RO-7). These readings are at the low end of the RO-7 instrument range; there are no subdivisions between 0 mR/hr and 10 mR/hr. CP readings were 25 mR/hr window open and 3 mR/hr window closed at the tunnel opening,

indicating presence of a low energy β emitter. See Appendix H for specific radiological survey data.

This area was surprisingly clean, despite its position as the lowest point directly underneath the reactor. The space underneath the moderator dump tank is drained directly to the previous location of the A-cell sump pump. The tunnel is a few inches off the floor, making evaporation the only way for the last part of any collected liquid to dry. Because of that, expectations were to find some dirt on the floor. Although this was the case, smearable contamination was low. The aluminum tank dunnage looked to be in excellent condition (see data in Table 1.0, IV.D).

5.0 CHARACTERIZATION SUMMARIES BY SURFACE TYPE

The surface area for each surface type is estimated by calculating areas of simple geometric shapes that approximate the surface shape(s). These calculations are included in Appendix G, along with basic assumptions. Section 5.5, "Data Summary Presentations" contains Table 2.0 and Table 3.0. Table 2.0 presents area calculations and technical smear field and laboratory results. Table 3.0 presents total curie estimates for the various surface types discussed here.

5.1 INSIDE SURFACES

Defined here as surfaces that did not contact ambient air during operation and are not readily accessible.

5.1.1 Moderator (Table 1, II.A, "Heavy Water Dump Tank")

Description: This surface consists of all the surfaces that contained or contacted the moderator water. This includes the inside of the moderator dump tank, the bottom gas lines and bottom drain line, the top drain lines, the top gas lines, the outside of the shroud tubes, the inside surface of the reactor vessel, the top plate, the top of the reflector vessel, the moderator inlet plenum, and the moderator dump space. All of these surfaces are aluminum alloy.

Estimated surface area: $2.78 \times 10^6 \text{ cm}^2$ ($2.99 \times 10^3 \text{ ft}^2$)

Sample results or inferred conclusions:

$$\text{Total } \alpha = 2.59 \times 10^{-4} \text{ } \mu\text{Ci}/100 \text{ cm}^2$$

$$\text{Total } \beta = 1.84 \times 10^{-3} \text{ } \mu\text{Ci}/100 \text{ cm}^2$$

$$^{60}\text{Co} = 2.73 \times 10^{-3} \text{ } \mu\text{Ci}/100 \text{ cm}^2$$

$$^{137}\text{Cs} = 2.04 \times 10^{-5} \text{ } \mu\text{Ci}/100 \text{ cm}^2$$

$$^{241}\text{Am} = 6.57 \times 10^{-5} \text{ } \mu\text{Ci}/100 \text{ cm}^2$$

Information source: Smears taken from the inside of the moderator dump tank. (It should be noted that the bottom of the moderator dump tank is likely the most contaminated part of this surface. Because of this, any curie content calculations will be higher than actual content).

5.1.2 Primary Coolant (Table 1.0, II.B, “Primary Piping Internals/Externals”)

Description: This surface consists of all the surfaces that contained or contacted primary coolant. This includes the inside of the inlet and outlet ring headers and their respective supply and drain lines, the jumpers between the ring headers and the process tubes, and the inside of the process tubes and nozzles.

Estimated surface area: $2.19 \times 10^6 \text{ cm}^2$ ($2.36 \times 10^3 \text{ ft}^2$)

Sample results or inferred conclusions:

$$\text{Total } \alpha = 2.12 \times 10^{-5} \text{ } \mu\text{Ci}/100 \text{ cm}^2$$

$$\text{Total } \beta = 2.37 \times 10^{-3} \text{ } \mu\text{Ci}/100 \text{ cm}^2$$

$$^{60}\text{Co} = 3.05 \times 10^{-3} \text{ } \mu\text{Ci}/100 \text{ cm}^2$$

$$^{137}\text{Cs} = 1.33 \times 10^{-6} \text{ } \mu\text{Ci}/100 \text{ cm}^2$$

Information source: Smears taken from the inside surface of the nozzles on tubes numbered 1550, 1354, and 1144.

5.1.3 Dry Gas Space Surface (Table 1.0, II.C, “Dry Gas Space”)

Description: This is the space between the reactor core socket formed by the biological shield and the calandria. It includes the inside of the biological shield, the outside of the calandria, down each shroud tube (between the shroud tubes and the process tubes), down through the bottom shield along the process tubes to the gas seals on the bottom, along the side of the bottom shield and top shield to the gas seals. Additional area includes the entire surface area of the thermal shield, out along the side step plug sockets and the outside surface of the thimbles.

Estimated surface area: $4.42 \times 10^6 \text{ cm}^2$ ($4.75 \times 10^3 \text{ ft}^2$)

Sample results or inferred conclusions: Neutron counts were below background levels, making the neutron analysis very sensitive. Because of the low counts, there is very little, if any, fissile material in the calandria. Gamma spectrum analysis could not yield further information on transuranic amounts because ^{60}Co and ^{137}Cs emissions were so high their Compton scatter masked spectra emitted by low-energy emitting isotopes, as the transuranics would be.

Information source: Gamma spectral analysis and neutron count through the thimble wall (Appendix H).

5.1.4 Reflector Surface

Description: This is the area inside the reflector tank and its supply and discharge piping.

Estimated surface area: $4.77 \times 10^5 \text{ cm}^2$ (513 ft²)

Sample results or inferred conclusions: This surface is assumed to be the same as the inside of the thimble. No isotopic data were obtained from the thimble technical smears. However, field treatment readings were 60,000 dpm β/γ and <20 dpm x.

Information source: Thimble IL technical smear field data.

5.1.5 Top and Bottom Shields

Description: The top and bottom shields are essentially mild steel tanks filled with iron shot. Depending on where the shot is located, it started out at either 0.1 in. or 0.25 in. in diameter. The most important question regarding these shields was whether they are dry or wet, because that would indicate the condition of the iron shot. Both top and bottom shields exceed the capacity of the polar crane with the shot in them so condition of the shot is important if the core is to be dismantled.

Estimated surface area: N/A

Sample results or inferred conclusions: The bottom shield is dry. This is known because the fill and drain lines have both been cut, the fill line came off the bottom of the tank, and the cut ends of the lines are both below the bottom shield. The top shield is also dry, according to experimental data.

Information source: This is inferred from data gathered from a PuBe/Cu probe (Appendix E). This probe consists of a PuBe radioactive source inserted into a copper pipe. The probe was lowered into a process tube to a level near the top of the top shield and also near the bottom of the top shield. The probe was also lowered to the bottom shield mid-plane to provide a "blank." The PuBe source emits fast neutrons which pass through the copper pipe and into the surrounding area. If the neutrons are thermalized and reflected back they will activate the copper. Readings taken at the two elevations in the top shield are low and essentially the same, leading to the conclusion that there is no water in the shield. They also correlated well with the reading from the bottom shield, known to be empty.

5.2 OUTSIDE SURFACES

Defined as those contacting ambient air and fairly easily accessible.

5.2.1 Top Face (Table 1.0, II.D, "Reactor Top Face")

Description: For the purposes of this report, the top face is defined as the space bounded by the top of the top shield, the bottom of the rotating and stationary shields, and the inside of the biological shield socket from the -2 ft level to the -5 ft level. This area contains the outlet ring header, outlet jumpers, and outlet nozzles.

Estimated surface area: $9.11 \times 10^5 \text{ cm}^2$ (969 ft²)

Sample results or inferred conclusions:

Total $\alpha = 8.23 \times 10^{-6} \text{ } \mu\text{Ci}/100 \text{ cm}^2$

Total $\beta = 3.09 \times 10^{-5} \text{ } \mu\text{Ci}/100 \text{ cm}^2$

⁶⁰Co = $1.65 \times 10^{-5} \text{ } \mu\text{Ci}/100 \text{ cm}^2$

¹³⁷Cs = $1.12 \times 10^{-5} \text{ } \mu\text{Ci}/100 \text{ cm}^2$

²⁴¹Am = $5.73 \times 10^{-6} \text{ } \mu\text{Ci}/100 \text{ cm}^2$

Information source: Smears and direct dose rate readings.

5.2.2 Bottom Face (Table 1, II.E, "Reactor Bottom Face")

Description: For the purposes of this report, the bottom face is defined as the space bounded by the bottom of the bottom shield, the top of the thermal barrier, and the cylindrical portion of the lower reactor space above the thermal barrier.

Estimated surface area: $1.16 \times 10^6 \text{ cm}^2$ ($1.25 \times 10^3 \text{ ft}^2$)

Sample results or inferred conclusions: Although smears were not available, it is reasonable to assume the bottom face to be similar to the lower reactor space. Using the piping smear data from the lower reactor space will give the most conservative estimate.

Information source: Lower reactor space contamination data.

5.2.3 Lower Reactor Space (Table 1, II.F, "Lower Access Space")

Description: This space is underneath the thermal barrier on the bottom of the reactor. It is closed off on the west by the walls of B-cell and C-cell and on the south by the labyrinth between C-cell and A-cell. The space opens into A-cell in the northeast quadrant, with a structural pillar dividing the opening into two parts. The openings into A-cell have been blocked off with temporary barriers made of light gage steel sheeting. The floor of this space is the top of the moderator dump tank and a catwalk surrounding the tank at the -32 ft level. This space was evaluated in more detail to support likely stabilization activity.

Estimated surface area:

Walls and Ceiling	$6.28 \times 10^5 \text{ cm}^2$ (676 ft ²)
Floor	$2.50 \times 10^5 \text{ cm}^2$ (269 ft ²)
Pipes	$1.70 \times 10^4 \text{ cm}^2$ (183 ft ²)

Sample results or inferred conclusions:

Walls and Ceiling	Total α = not detectable
	Total β = not detectable
	^{137}Cs = $3.14 \times 10^{-6} \text{ } \mu\text{Ci}/100 \text{ cm}^2$
Floor	Total α = not detectable
	Total β = $1.15 \times 10^{-4} \text{ } \mu\text{Ci}/100 \text{ cm}^2$
	^{60}Co = $5.70 \times 10^{-6} \text{ } \mu\text{Ci}/100 \text{ cm}^2$
	^{137}Cs = $6.28 \times 10^{-5} \text{ } \mu\text{Ci}/100 \text{ cm}^2$
Pipes:	^{241}Am = $1.99 \times 10^{-6} \text{ } \mu\text{Ci}/100 \text{ cm}^2$
	Total α = $6.65 \times 10^{-6} \text{ } \mu\text{Ci}/100 \text{ cm}^2$
	Total β = $1.13 \times 10^{-3} \text{ } \mu\text{Ci}/100 \text{ cm}^2$
	^{60}Co = $3.95 \times 10^{-6} \text{ } \mu\text{Ci}/100 \text{ cm}^2$
	^{137}Cs = $4.66 \times 10^{-4} \text{ } \mu\text{Ci}/100 \text{ cm}^2$
	^{241}Am = $3.94 \times 10^{-6} \text{ } \mu\text{Ci}/100 \text{ cm}^2$

Information source: Direct dose readings, gross and technical smears.

5.2.4 Beneath Moderator Dump Tank (Table 1.0, II.G, "Space Under Dump Tank")

Description: This is nearly the lowest place in the reactor building. It is a crawl space underneath the moderator dump tank. The moderator dump tank was constructed with a sump or well in the northeastern part section. To accommodate the sump, the tank was placed on dunnage that lifts the tank about 0.9 m (3 ft) off the floor.

Estimated surface area: $6.61 \times 10^5 \text{ cm}^2$ (712 ft²)

Sample results or inferred conclusions:

Total α = $7.07 \times 10^{-4} \text{ } \mu\text{Ci}/100 \text{ cm}^2$
Total β = $2.21 \times 10^{-3} \text{ } \mu\text{Ci}/100 \text{ cm}^2$
^{60}Co = $1.70 \times 10^{-5} \text{ } \mu\text{Ci}/100 \text{ cm}^2$
^{137}Cs = $2.05 \times 10^{-3} \text{ } \mu\text{Ci}/100 \text{ cm}^2$
^{241}Am = $1.01 \times 10^{-4} \text{ } \mu\text{Ci}/100 \text{ cm}^2$

Information source: Video, direct dose readings, gross and technical smears.

5.3 DOSE RATE SUMMARY

Dose rates inside the core (Table 1.0, III.A) range from a high of 44.3 R/hr in the center tube immediately below the bottom plate of the top shield to 60 mR/hr in a fringe tube near the

top of the top shield. The dose rate profile generated by this activity for tube number 1354 (Figure 2.0) can be compared to dose rates on a 1970 (Appendix H) survey of tube number 1350 because they have pretty much the same position relative to the center tube, number 1550. The 1970 (Appendix H) survey was made by removing the tube and measuring dose through shielding. Two surveys report this activity and can be found in Appendix H. When comparing dose rates, take into account the fact that the 1970 (Appendix H) survey removed the tube and the 1997 survey (Appendix H) took dose rates down through the tube and includes background from other tubes, thermal shield, and top and bottom shield.

Dose rates through core shielding (Table 1.0, III.B) are minimal. The biological shield provides complete protection from any dose generated in the reactor core. With the rotating shields in place, the dose rate on top of the reactor is also zero. Direct contact with the process tube nozzles (rotating shields removed) gives a high dose of 0.5 to 2 mR/hr on contact with the nozzle caps in place. This dose falls off rapidly and yields no whole body dose to a person standing on top of the nozzles.

Field dose rates in the lower reactor space range from 10 mR/hr 15 cm from the moderator dump tank access plate and 8 mR/hr at one place in the ceiling to a low of 1.5 mR/hr in the southern-most point of the room. General field readings are about 2.5 mR/hr.

Table 4, in Section 5.5 "Data Summary Presentations," presents all core dose profile information. Table 5.0 presents dose rate information through the biological shield. Figure 3.0 plots dose readings from Tables 4.0 and 5.0 and plots them against a scale drawing of the reactor core and biological shield. The readings on the left in the horizontal graph were taken through the biological shield in the 1L thimble. Readings from the proper elevation (-156 in.) were included to show a complete dose profile from outside of the biological shield to the centerline of the reactor core. The sharp increase in dose rate near the inside edge of the biological shield is expected because of dose rate "shine" as the probe left the biological shield and passed through the cast iron thermal shield. Cobalt impurity in the cast iron thermal shield was activated during reactor operations and accounts for the rapid rise in dose. Because of self-shielding, the dose rates on the inside of the thermal shield are higher than dose rates in the middle of its mass. The second bump in the graph can be accounted for by activation in the aluminum vessel wall, and the rise in dose rates into the core centerline can be accounted for by activation of the zircalloy process tubes and the aluminum shroud tubes inside the core and cumulative dose effects.

The readings on the graph to the right depict dose rates as a function of 1) elevation down through the reactor and 2) distance from the core centerline. As expected, activation of the mild steel bottom plate of the top shield yielded the highest dose. This is expected because during those times when the reactor was not up to full power, or the moderator was at a level less than completely at the top of the vessel, the neutrons at the top of the reactor would have been bouncing between the moderator surface, the light water in the top shield, and the annular reflector tank, making this area high in neutron density. The dose rate peak near the bottom of the reactor, while lower than the top shield peak, exists for the same reason as the top shield peak. The broad peak around the middle of the reactor core height is probably caused by the activation of the zircalloy process tubes, and reflects the shape of the neutron field during operation. Combining these three peaks results in the dose rate profile shown in the figure. There are three

Figure 3.0. Dose Rate Profiles.

lines on the graph that mirror each other. The line with the highest peak represents data taken in the center tube, number 1550. The next lower line represents data taken from tube number 1354, mid-way between the core centerline and the vessel wall. The smallest line represents data taken from a fringe tube number 1144. These tubes are not in line with each other, so the assumption was made that metals activation would be symmetrical around the core centerline.

5.4 RADIOCHEMISTRY RESULTS

Radiochemistry was used to determine which isotopes are responsible for giving off the dose rates and to predict the drop in dose over time. This was done using in situ gamma energy spectral analysis. Also, radiochemistry was used to determine the greatest amount of plutonium or other fissile material that may be in the dry gas space as a legacy of the 1965 accident. This was done by combining neutron readings through the aluminum thimble wall with gamma energy spectral analysis.

Gamma energy spectral analyses indicate most of the dose comes from ^{60}Co and ^{137}Cs (see Appendix I). The radiation from these isotopes is so strong that other isotopes, in smaller quantities, are masked. Neutron counting through the thimble indicate the maximum amount of plutonium or other fissile material inside the dry gas space to be 0.92 grams.

Appendix J contains calculations predicting future dose rates and isotopes based on standard radioactive decay schemes for prominent isotopes.

5.5 DATA SUMMARY PRESENTATIONS

Tables 2.0, 3.0, 4.0, and 5.0 provides a summary of data analyzed.

Table 2.0. Contamination Data Summary Table. (2 sheets)

Surface	Estimated area (cm ²)	Field data (dpm/100 cm ²) ¹		Laboratory data (Appendix H) (μCi/100 cm ²)		
		α	β & γ	Isotopic amounts	Total α	Total β
Moderator	2.78 x10 ⁶	700	15,000	$^{60}\text{Co} = 2.73 \times 10^{-3}$ $^{137}\text{Cs} = 2.04 \times 10^{-5}$ $^{241}\text{Am} = 6.57 \times 10^{-5}$	2.59x10 ⁻⁴	1,840 x10 ⁻³
Primary	2.19 x10 ⁶	<20	6,330	$^{60}\text{Co} = 3.05 \times 10^{-3}$ $^{137}\text{Cs} = 1.33 \times 10^{-6}$	2.12 x10 ⁻⁵	2.37 x10 ⁻³
Dry gas ²	4.42 x10 ⁶					
Bottom face ²	1.16 x10 ⁶					
Reflector ³	4.77 x10 ⁵	<20	60,000			

Table 2.0. Contamination Data Summary Table. (2 sheets)

Surface	Estimated area (cm ²)	Field data (dpm/100 cm ²) ¹		Laboratory data (Appendix H) (μCi/100 cm ²)		
		α	β & γ	Isotopic amounts	Total α	Total β
Top face	9.11 x10 ⁵	<20	611	⁶⁰ Co = 1.65 x10 ⁻⁵ ¹³⁷ Cs = 1.12 x10 ⁻⁵ ²⁴¹ Am = 5.73 x10 ⁻⁶	8.23 x10 ⁻⁶	3.09 x10 ⁻⁵
Lower reactor space						
Walls/ceiling	6.28 x10 ⁵	<20	1,000	¹³⁷ Cs = 3.14 x10 ⁻⁶	<9 x10 ⁻⁶	<5 x10 ⁻⁵
Floor	2.50 x10 ⁵	<20	2,000	⁶⁰ Co = 5.70 x10 ⁻⁶ ¹³⁷ Cs = 6.28 x10 ⁻⁵ ²⁴¹ Am = 1.99 x10 ⁻⁶	<1 x10 ⁻⁵	1.15 x10 ⁻⁴
Pipes	1.70 x10 ⁵	<20	8,500	⁶⁰ Co = 3.95 x10 ⁻⁶ ¹³⁷ Cs = 4.66 x10 ⁻⁴ ²⁴¹ Am = 3.94 x10 ⁻⁶	6.65 x10 ⁻⁶	1.13 x10 ⁻³
Under moderator dump tank	6.61 x10 ⁵	1.75 E2	12,300	Co ⁶⁰ = 1.70 x10 ⁻⁵ Cs ¹³⁷ = 2.05 x10 ⁻³ Am ²⁴¹ = 1.01 x10 ⁻⁴	7.07 x10 ⁻⁴	2.21 x10 ⁻³

¹Table reports field instrument data in disintegrations per minute (dpm). Numbers reported above in the text for total α and total β are for laboratory instruments and reported in μCi per sample or μCi/100 cm².

²Contamination data not available for this surface.

³Isotopic data not available for this surface.

Table 3.0. Total Curie Estimates by Surface Type.

Surface	Estimated area (cm ²)	Estimated curie contamination of surface type (μCi)				
		⁶⁰ Co	¹³⁷ Cs	²⁴¹ Am	Total α	Total β
Moderator	2.78 x10 ⁶	7.59 x10 ¹	5.67 x10 ⁻¹	1.83 x10 ⁰	7.20 x10 ⁰	5.12 x10 ¹
Primary	2.19 x10 ⁶	6.68 x10 ¹	2.91 x10 ⁻²	N/A	4.64 x10 ⁻¹	5.19 x10 ¹
Top face	9.11 x10 ⁵	1.50 x10 ⁻¹	1.02 x10 ⁻¹	5.22 x10 ⁻²	7.50 x10 ⁻²	2.81 x10 ⁻¹
Lower reactor space						
Walls/ceiling	6.28 x10 ⁵	N/A	1.97 x10 ⁻²	N/A	N/A	N/A
Floor	2.50 x10 ⁵	1.43 x10 ⁻²	1.57 x10 ⁻¹	4.98 x10 ⁻³	N/A	2.88 x10 ⁻¹
Pipes	1.70 x10 ⁵	6.72 x10 ⁻³	7.92 x10 ⁻¹	6.70 x10 ⁻³	1.13 x10 ⁻²	1.92 x10 ⁰
Under moderator dump tank	6.61 x10 ⁵	1.12 x10 ⁻¹	1.36 x10 ¹	6.68 x10 ⁻¹	4.67 x10 ⁰	1.46 x10 ¹

Table 4.0. Core Dose Rate Profile.

	Level	Tube			Tube		Tube	
		1550			1354		1144	
	Cutie Pie (CP) Readings							
	0 (in.)	3.5 mR/hr			3.7 mR/hr		3.5 mR/hr	
	R0-7 Readings (mR/hr)							
	Level (in.)	Low range	Mid-range	Mid-range	Low range	Mid-range/hr	Low range	Mid-range/hr
	0	3	0		3	0	4	0
Top shield	-24	10	0		12	0	11	0
	-44	92	200	300	89	200	53	100
	-48			600				
	-52			1,800				
	-56			8,800				
	-58			15,600				
	-60			23,900				
	-62			32,800				
	-64		38,500	40,300		34,900		19,300
-66		46,700	44,300		42,200		26,000	
Reactor Vessel	-68			28,100				
	-70			22,000				
	-72		19,700	19,500		16,100		12,900
	-84		14,600			13,000		11,400
	-96		15,400			13,500		12,300
	-108		16,800			14,700		13,300
	-120		17,100			14,900		13,600
	-132		15,900			14,000		12,800
	-144		13,800			12,500		11,000
	-156		11,600			11,000		9,300
	-168		10,100			9,300		8,100
-180		13,500			11,200		7,600	
Bottom shield	-192	1,975	700		1,245	600	1,000	200
	-204	30	0		28	0	15	0

Table 5.0. Dose Rates Through Biological Shield up to Reactor Vessel Wall.

Distance (in.)	Teletector (mR/hr)	RO-7, low range (mR/hr)	RO-7, high range (mR/hr)
0	4	3	0
6	5.5	4	0
12	6.5	7	0
18	8	8	0
24	9	9	0
30	15	11	0
36	20	14	0
42	28	22	0
48	40	33	0
54	65	52	100
60	160	95	200
66	320	205	300
72	1,000	625	1,300
78	2,500	1,000	4,700
84	8,000	off scale	11,100
90	8,500	--	9,500
96	8,500	--	9,300
102	8,500	--	9,700
106	8,600	--	9,700

5.6 CONCLUSION AND RECOMMENDATIONS

Because of the relatively benign conditions found in the PRTR reactor core and supporting structures, this report recommends that stabilization action be taken only in the lower reactor space. Clean-up of the lower reactor space will generate low-level waste that contains asbestos. Following this stabilization effort, the reactor dome should be maintained for about 20 to 30 yr. At the end of a 20 yr period, radiological conditions (estimated at 1.2 R/hr) should be low enough for relatively easy dismantling of the building. Presently, contamination in the PRTR reactor core is not migrating, and has no way to do so as long as the reactor dome is kept dry. Of course, exposing contamination areas to flowing water would permit contamination to become mobile. Dose rates and contamination levels in the PRTR reactor core are low, and will only decrease with time.

During access to the PRTR at the 21-ft level, the fission chamber located in the -14.6 level was removed and disposed of as an expedited cleanup item.

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6.0 APPENDIX

- Appendix A. PRTR Core Information Summary and Analysis
- Appendix B. PRTR Lead Survey
- Appendix C. Pictures of Characterization Work
- Appendix D. Work Plan
- Appendix E. Radio-Chemistry Report on Water in the Top Shield
- Appendix F. Radiological Surveys and Laboratory Data
- Appendix G. Surface Area Calculations and Assumptions
- Appendix H. Old Surveys
- Appendix I. Radio-Chemistry Report on Gamma and Neutron Data
- Appendix J. Dose Rate Prediction Calculations

APPENDIX A

PRTR CORE INFORMATION SUMMARY AND ANALYSIS

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309 BUILDING

ADVANCED FUEL FACILITIES TRANSITION

PRTR Core Information Summary and Analysis

**By Nolan Hale, ASI
for Westinghouse Hanford Company**

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PRTR Core Information Summary and Analysis

PURPOSE

The purpose of this document is to provide a “thumb-nail sketch” of the PRTR core for the person(s) assigned to core characterization and subsequent decommissioning. It is a brief overview and contains references to original documents and drawings from which this information was gleaned.

CONSTRUCTION MATERIALS

The materials used in constructing the PRTR reactor and shielding are:

System	Materials
Calandria	Aluminum alloys (6061, 6063, 5050, 5052)
Reflector	Aluminum alloys (6061, 6063, 5050, 5052)
Top and Bottom Shields	“Mild Steel” tank filled with iron shot
Thermal Shield	Cast Iron
Biological Shield	High density concrete with a mild steel jacket
Process Tubes	Zircalloy 2
Rotating Shields	Reinforced Concrete with steel facings
Inlet and Outlet Headers	Stainless Steel
Inlet and Outlet Jumpers	Stainless Steel
Moderator Dump Tank	Aluminum Alloy
Piping Between Dump Tank and Calandria	Aluminum Alloy
Shims	Inconel
Thermal Barrier	MgO powder sandwiched between Al plates

CONFIGURATION

Core configuration is presented in operating systems for three reasons: 1) equipment configuration depends on its function, 2) it is easier to visualize how systems work together than unrelated components, and 3) system operations have bearing on probable locations of contamination. Refer to Figure 1 for information on spatial relationships between components.

PRTR Core Information Summary and Analysis

PRTR Core Information Summary and Analysis

Moderator System

The PRTR was controlled by moderator level. When the moderator level was highest, the reactor was at full power. Lower power levels were achieved by lowering the moderator level, thereby reducing the number of thermal neutrons available for the reaction.

Moderator effectiveness was influenced by the “shim rod” system and the “chemical shim” system. Shim rods were positioned in various parts of the neutron flux to shape it. The shim rods were made of Inconel and acted as neutron “drains” to capture neutrons, reducing the number of thermal neutrons available for reaction. A chemical shim system was added after a September 1965 accident. The chemical shim worked by spiking the moderator with boron prior to startup. Boron captures neutrons and poisons the reactor. Boron was stripped out of the moderator by an ion exchange unit to maintain constant reactivity for a given moderator level as fuel burned up.

The PRTR was “scrammed” by dumping all the moderator out of the reactor. The moderator system consists of the moderator dump tank, pump and cooling system, inlet plenum, moderator plenum, dump chamber, moderator vessel, level control equipment, and connective piping.

- The moderator dump tank is an aluminum tank placed approximately 11 feet below the reactor core. It is large enough to contain the entire volume of reactor moderator water. It is connected to the top of the reactor by four top gas lines and two different drain lines. The top gas lines are 10" pipes with a dump valve that isolated the gas above the reactor from the gas in the dump tank when closed. The top drain lines run from the bottom of the tank sump to the top of the reactor so it was water-filled during reactor operation. The dump tank was connected to the bottom of the reactor (the dump chamber) by four bottom gas lines and one bottom drain line. The moderator pump system took suction from the bottom of the dump tank sump.
- The moderator pump system moved moderator from the dump tank sump, through a cooler, and into the inlet plenum. This entire system has been removed except for the in-reactor piping and is of no further interest here.
- The inlet plenum is a chamber bounded by the calandria base plate on the bottom, the moderator plenum plate on top, and a cylindrical wall joining the two. Moderator water was pumped into the inlet plenum in two locations and the water escaped into the moderator plenum by passing through holes in the plenum plate.
- Atop the inlet plenum is the moderator plenum. It is formed with the plenum plate on the bottom and an annular rim welded around the plenum plate to form a deep dish.
- The moderator dump chamber surrounds the inlet plenum and drained to the moderator dump tank through the drain line (and the bottom gas lines during a scram). The moderator dump chamber could contain enough moderator to stop the reaction, although there would still be moderator up in the moderator vessel.

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Figure 3 Calandria Functional Model

- The moderator vessel is an open bottomed tank, with the plane of its lower, open end below the plane described by the top of the moderator plenum. As water rose in the moderator plenum it would form a seal between the gas in the moderator vessel and the moderator dump tank. This seal completely isolated these two gas volumes from each other, allowing the moderator level control system to work.

Moderator level was controlled by creating a low pressure in the moderator vessel and a high pressure in the moderator dump chamber (same gas space as in the dump tank). The moderator vessel functioned much like a drinking glass filled with water and inverted over a water-filled saucer. The column of water in the glass is held up by the difference in air pressure above the column of water and the air pressure above the free-surface water in the saucer (Figure 2). The only functional difference between the moderator level system and this model is that the moderator level system could vary pressure differential and actually drew water up into the moderator vessel. A blower moved helium from the moderator vessel to the dump tank around one of the dump valves. Fine control of the moderator level was achieved by five different control valves that could bypass the blower, making it less effective. Although moderator water was pumped into the moderator inlet plenum, water level was controlled by gas pressure differential. Excess water spilled over from the moderator plenum to the dump tank via the moderator dump chamber or the moderator drain system.

The reactor was “scrammed” by opening the four dump valves in the top gas lines and equalizing pressure above the moderator vessel and in the moderator dump tank. Moderator water gravity drained from the moderator vessel, taking the reactor from full power to sub-critical (a two-foot drop in moderator level) in .85-.95 seconds.

Reflector Tank

The neutron reflector chamber is an annular tank surrounding the moderator chamber and using a common wall. The neutron reflector has ten penetrations that allow access to the wall of the

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moderator chamber from outside the reactor. These penetrations allowed placing instrumentation in the neutron flux while the reactor was critical.

Shield System

The PRTR had several shield systems: biological shield, top and bottom primary shields, thermal shield, rotating shields, the top sheet cooling system, and the thermal barrier.

- The biological shield provides structural support for the entire reactor core, as well as neutron attenuation. Its center forms a “socket” into which the reactor core is placed. The inside of the socket is lined with mild steel and has a cooling system. The concrete is a high density concrete that attenuated thermal neutrons created during reactor operations. The bottom primary shield rests on the bottom of the biological shield socket and the top primary shield rests on a ledge near the top of the biological shield.
- The bottom primary shield is a mild steel tank (weighing 12.2 tons) filled with 11.1 tons of iron shot. When the shield’s cooling water was added, the total weight of the shield was 44.8 tons. The shield has perforations to allow for fuel process tubes, moderator inlet, top and bottom gas lines, and other services. Shield plugs were manufactured and inserted in the perforations to make a tight fit between the piping and the shield perforations, thereby stopping radiation leaks from the core. The calandria sits on top of the bottom shield.
- The top primary shield is above the calandria. Its construction is similar to the bottom primary shield, with iron shot filling and a method for circulating water for cooling. It is thicker in the middle than at the edges to accommodate the toroid-shaped outlet header that surrounds it. The top shield without shot and water weighs 19.1 tons. It contains 55.5 tons of shot and an additional 26.8 tons of cooling water that circulated through it. The top primary shield also has piping and shield plugs that allow for “through reactor” systems and various services.
- Surrounding the reflector is the thermal shield. It is made of cast iron plates fitted together. The thermal shield has its own cooling system.
 - The rotating shields are designed to provide access to any of the process or other tubes, one at a time. This system limited radiological exposure by uncovering only the part of the reactor needing attention, while all the other parts of the reactor top remained shielded. The rotating shield system consists of an outer stationary shield ring with two, circular, inner rotating shields (Figure 3). The inner rotating shield sits eccentrically within the outer rotating shield and has an eccentrically located shield plug which provided access to the reactor top. The rotating shields each sit on a ball bearing race and are moved by turning a pinion gear against an outer gear. The entire shield system is designed with an interlocking flange system that locks the shields in place when in the “zero,” or operating, position. This system is like a bayonet system to lock an interchangeable lens on a camera body. This was

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a safety feature to keep the shields from blowing out in case of accident. Inflatable seals between the rotating shields provided a gas seal barrier between breathable air and air on top of the reactor.

- The top sheet cooling system was not technically a shield, however, it served the purpose of removing excess heat that would otherwise escape from the top of the reactor. The top sheet cooling system was a void between the calandria top sheet and the calandria top plate. This void had heavy water pumped to it from a branch line off the moderator inlet piping. The water streamed over the top sheet and drained into the overflow drain system.
- The thermal barrier is also not technically a shield. It is a barrier made of MgO powder sandwiched between aluminum plates. The thermal shield serves as the “ceiling” for the lower reactor space and consists of a fixed portion and a moveable portion. The moveable portion could be lowered to provide access to the inlet ring header, jumpers, angle valves, and other features on the reactor bottom face.

Figure 4 Rotating Shield System

“Through Reactor” Systems

The calandria has three types of tubes running through it: fuel shroud tubes (85), shim tubes (18), and flux monitoring tubes (13). These tubes have no gas connection to the water side of the calandria and were kept under a blanket of dry helium.

- Fuel shroud tubes have pressure tubes running through them, from the bottom of the bottom primary shield up through the top shield. The bottom of the pressure tubes are connected to the primary coolant inlet ring header (a toroid-shaped header under the reactor) via “jumpers.” On top each pressure tube is an outlet nozzle that comes to about 6" below the rotating shields. The nozzles are connected to the outlet ring header via “jumpers.” The primary coolant outlet ring header is a toroid-shaped header that lays around the top primary shield. Dry helium filled the space between the outside of each pressure tube and the inside of its fuel shroud tube. Between the flow of the primary coolant system and the helium blanket, enough heat was removed from the reactor fuel elements that they could be very hot (near or at melting point for ceramic fuel) while the moderator water was not hot enough to boil.

Primary coolant flowed into the reactor core through the inlet header on the bottom of the

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reactor, to the pressure tubes via inlet jumpers, through the pressure tubes containing nuclear fuel, out the pressure tube nozzles on top of the reactor, through outlet jumpers to the outlet header and off to the cooling and pumping parts of the system.

Fuel rod clusters hung down in the pressure tubes from a hanger system in the nozzle. Since all fuel rods have been removed, they are not treated in this paper.

- Shim tubes have a smaller diameter than fuel shroud tubes. They each contain three Inconel shims that travel up and down the height of the moderator vessel on long, aluminum screws. The shims are each 36" long with a pie-shaped cross section. Shim range of motion extends from within the top shield to within the bottom shield.
- Flux monitoring tubes extend only to the moderator plenum plate. Only six of the thirteen flux monitoring tubes were used. Flux was monitored by inserting a wire into the reactor, letting it become activated, and then removing it while reading radiation energy levels.

ACCESS TO THE CORE

Present day access to the core is limited because of the removal of equipment and grating that were designed to facilitate core access. However, there are several ways to gain direct contact with reactor core materials to get information regarding contamination levels and overall dose rates.

Reactor Top Face

The top of the reactor can be accessed through two different means. The first is to lift out the rotating shield access plug. This would allow radiological smears to be taken from the bottom of the plug as well as the top face, a few outlet nozzles, and to determine general dose rates. The second way is to remove one of two shield plugs located in "C" cell on the -11' level. This would allow shielded access to take a radiological smear from the bottom of the rotating shields and take general dose rates, but would not allow direct contact with the top face. There are also about six access plugs to the moat, but they would not allow the contact needed to make a fair assessment of the reactor top.

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Calandria

The best access to the calandria is through one of the test or flux chamber holes in the "C" cell. Removal of the spiral step plug will allow access to the side of the moderator vessel. Radiological smears could be taken from the thermal shield, the outside of the calandria (reflector tank), and the moderator vessel. Direct radiation readings could be taken from several places in the access: thermal shield, inside the reflector tank, and on the moderator vessel wall. It is also possible to drill a hole in the moderator vessel wall and take a reading directly against one of the fuel shroud tubes. Since there is 1.5-2" between the thermal shield and the entrance into the calandria it would be possible to videotape the side of the calandria down to the top of the bottom shield or up to the gas supply header. It would also be possible to break lines into the calandria (top gas line, bottom gas line) and run instrumentation directly into different parts of the calandria. This might yield information regarding crud development in the moderator vessel or deposits in the top drain space.

Bottom Face

The bottom face of the reactor is the most easily accessed. On decommissioning, the reactor crew fashioned some "temporary" barriers out of sheet metal and bolted them over the openings to the lower access room. These could be removed, allowing free access to the bottom face and the top of the moderator dump tank. Research indicates the moveable portion of the thermal barrier is lowered.

PROBABLE CONTAMINATION LOCATIONS

Based on a report¹ of the September 29, 1965 Class A accident, it is quite possible that a portion of the lost fuel is still on top of the reactor and down in the thermal shield. This report indicates that the fuel aerosol invaded the dry helium space in the reactor. If true, the contaminated steam would have condensed and plated out fuel and fission products throughout the gas space of the reactor core. This will play a major part in dismantling the reactor. Other places likely contaminated from the accident are the biological shield access holes and inside shroud tubes.

Contamination from normal operations will likely be seen in the moat and around the outside of the top gas seal. At a minimum, contamination would concentrate here because of the few times a fuel element was sprayed with cooling water while in the refueling machine. Also, the moderator dump tank has never been opened since it was put into service. It is possible, though unlikely, that some heavy water still exists in the sump. At any rate, the inside of the tank is likely contaminate. The moderator dump tank sits on dunnage in the dump tank pit. No record has been found documenting the contamination levels underneath the moderator dump tank and it is likely quite contaminated.

¹NUREG/CR-3669, *Plutonium Recycle Test Reactor (PRTR) Accident, A Final Report on the Investigation of Fission Product Chemical Forms*, W.K. Hensley and L.A. Rogers, April 1984.

PRTR Core Information Summary and Analysis

CHARACTERIZATION STEPS

1. Determine present radiological conditions.
2. Account for any hazardous materials (most notably lead).
3. Identify long-lived activation products.

DISMANTLING PROCESS

1. Remove rotating shields (H-3-11540)
 - a. Remove caulk or grout from between shields, lifting bolt sockets, and rotation machinery parts
 - b. Remove rotating shield step plug (lift slightly and rotate until the bayonet flanges allow for removal; 0.41 tons)
 - c. Rotate inner disc to unlocked position and remove (6.4 tons)
 - d. Rotate outer disc to unlocked position and remove (31 tons)
 - e. Remove top stationary ring (17 tons)
 - f. Remove bottom stationary ring (19 tons)
2. Remove all in-tube equipment
 - a. Remove fuel pressure tubes
 - i. Disconnect and remove all outlet jumpers
 - ii. Disconnect and remove all inlet jumpers (bottom of reactor) *(leave the angle valves on the ring header)*
 - iii. Remove all outlet nozzles
 - iv. Remove and discard all pressure tubes *(use H-3-11360 to design lifting mechanism for removing pressure tubes)*
 - v. Remove pressure tube sampling lines. Remove to bulkhead fittings and cap bulkhead fittings.
 - b. Disconnect and remove all shim control equipment
 - c. Disconnect and remove all flux monitoring equipment
 - d. Disconnect and remove any cables or other equipment left on top of the top primary shield (RTD's, control and instrument wiring, etc.)
3. Remove top primary shield cooling water lines
4. Remove outlet header
 - a. Cut or remove expansion joint from the outlet header discharge pipe
 - b. Lift out the outlet header

Note: the top and bottom primary shields far outweigh the capacity of the polar crane. The polar crane can be used to remove these structures only if emptied of iron shot and water (if any remains). Otherwise, the only option in removing the shields is with an external crane, possibly requiring removal of the top of the dome.

5. Remove the top primary shield
 - a. Remove process tube shield plugs
 - b. Remove iron shot. *Evaluate the status of the iron shot. If the shot are not rusted together it may be possible to cut accesses into the top shield and vacuum out the shot. In the event they are rusted together, it may be possible to separate them*

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with a mild acid solution (acetic or dilute nitric), or it may be necessary to cut the mass out with a torch or by other means.

- c. Cut top gas seal
- d. Remove top primary shield (expect fission product contamination on the bottom of the top primary shield)
- 6. Disconnect all calandria piping on the bottom of the reactor
 - a. Lower header drain line
 - b. Top gas lines (4)
 - c. Bottom gas lines (4)
 - d. Bottom drain line
 - e. Reflector drain line
 - f. Reflector supply line
 - g. Top tube sheet cooling line
 - h. Moderator inlet (2)
 - i. Upper header drain line (moderator)
 - j. Lower header drain line (moderator)
 - k. Dry gas lines
- 7. Remove dry gas supply header
- 8. Remove spiral step plugs and thimbles
 - a. Fabricate a new step plug removal handle (H-3-11519) and lifting bolts (H-3-11517).
 - b. Remove the step plugs
 - c. Remove the thimble from each access hole (expect fission product contamination)
- 9. Remove calandria (expect fission product contamination)
- 10. Remove thermal shield
 - a. Disconnect and remove cooling water inlet and outlet headers
 - b. Remove cooling lines
 - c. Remove thermal shield plates (expect fission product contamination)
- 11. Remove dry gas exhaust header
- 12. Remove bottom primary shield
 - a. Remove shield plugs
 - b. Remove shot mass from shield. (See discussion for shot in top shield)
 - c. Disconnect thermal barrier connections
 - d. Remove bottom primary shield
- 13. Remove inlet ring header
 - a. Cut remains of inlet jumpers back to angle valves
 - b. Cut the inlet header (it is too large to remove intact)
 - c. Remove the inlet header pieces
- 14. Remove thermal barrier
 - a. Disassemble the moveable portion of the thermal barrier and remove
 - b. Remove the stationary portion of the thermal barrier (cut out)
 - c. Cut the supports to the "stationary islands" around two of the pipes

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15. Remove moderator dump tank
 - a. Cut moderator pump header supply line at dump tank sump
 - b. Remove dump tank

Problems:

Expect contamination everywhere from the time the rotating shield plug is pulled. Most of the crane lifts in this process will remove equipment that has not been accessible since the reactor was built and could not possibly have been decontaminated.

The shot in the top and bottom shields may be rusted together. This is a contingency that needs to be planned for. Also, no information has been found to date regarding any draining of the top and bottom shields. The bottom shield would have been relatively easy to drain by simply opening the cooling water inlet line. The top shield would have been rather difficult to drain because of limited accessibility and the need to pump the water out of it. Any water left in these shields will have activated rust products in it.

Assuming that the iron in the top and bottom shields have rusted, expect the shields to have potentially explosive levels of hydrogen gas (a by-product of water being the oxygen source for the oxidation process). The shields should be well vented prior to doing any hot work or any work that may cause a spark inside the shield.

The alloy metals will certainly be activated. Although there has been considerable time for decay, expect radiation levels to be an ALARA concern (especially from the biological shield liner, the Zircalloy pressure tubes, and stainless steel or mild steel exposed to neutron flux).

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REFERENCES:

Documents:

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HWS-6779, *Procurement Specification for Rotating Shield Actuator, Plutonium Recycle Test Reactor, 309 Building*, General Electric Company, October 12, 1959.

HWS-6213, *Procurement Specification for Heavy Water Storage Tank, Plutonium Recycle Test Reactor, 309 Building*, General Electric Company, May 8, 1958.

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HW-50700, *The Plutonium Recycle Program - A Resume of the Concept, Program, and Facilities*, The Staff of the Advanced Engineering Operation Reactor and Fuels Research and Development Operation, June 12, 1957.

HW-75557, *Decontamination of the Plutonium Recycle Test Reactor*, J. A. Ayers, et.al., September 30, 1963.

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Drawings:

Calandria Drawings

H-3-11301	Reactor Assembly
H-3-11320	Calandria Tank Assembly
H-3-11321	Calandria Section Thru Top Gas Header
H-3-11322	Calandria Section Thru Plenums, Header & Core

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H-3-11325	Calandria Moderator Plenum Assembly
H-3-18489?	Top Shield Gas Seal
H-3-18600	Calandria Top Plate, Shroud Tubes, Funnels & Rail
H-3-18601	Calandria Bottom Plate Details
H-3-18602	Calandria Bottom Plate w/ Moderator Plenum
H-3-18603	Calandria Dump Sheet, Baffle, & Rod Weldment
H-3-18604	Calandria Base & Plenum Plate
H-3-18605	Calandria Sub-Assembly Inner Shell Header Reflector Ring
H-3-18606	Calandria Bottom Plate, Plenum, Inner & Outer Shells
H-3-18607	Calandria Coolant Outlets & Flux Chamber Weldment
H-3-18608	Calandria Sensing Tubes, Overflow Plate & Thermocouple Well
H-3-18609	Calandria Top Plate, Shroud Tubes, Funnels & Rail
H-3-18610	Calandria Tank Assembly
H-3-18611	Calandria Tank Assembly Bottom View
H-3-18613	Calandria Weir Baffle Plate (lower)
H-3-18616	Calandria Inner Shell Top
H-3-18634	Calandria Top Ring
H-3-18636	Calandria Plenum Plate, Support Ring & Weir
H-3-18640	Reactor Tube Index System
H-3-18683	As Built Diameters of Shim Control Tubes at Moderator Plenum Plate
H-3-18685	Calandria Plenum Support Ring
H-3-18687	Calandria Moderator Plenum & Bottom Plate Assembly
H-3-18695	Calandria Outside Diameters & Elevations As Built
H-3-18696	Calandria Inside Shell Final Inside Diameters & Calandria & Shield Final Heights

Thermal Shield

H-3-11471	Thermal Shield Plan & Sections
H-3-11473	Thermal Shield Coolant Header Assembly

Bottom Primary Shield

H-3-18656	Bottom Primary Shield Weldment Sub-Assembly
H-3-18657	Bottom Primary Shield Sub-Weldment
H-3-18658	Bottom Primary Shield Sections for DWG H-3-18658
H-3-18658	Bottom Primary Shield Assembly
H-3-18664	Bottom Primary Shield Upper Sector Plate

Top Primary Shield

H-3-18464	Top Primary Shield Sub-Assembly Lower Section
H-3-18645	Top Primary Shield Base Plate Detail
H-3-18647	Top Primary Shield Sub-Assembly
H-3-18648	Top Primary Shield Assembly
H-3-18649	Primary Shields, Sections for H-3-18648 & 58
H-3-18654	Top Primary Shield Water Distribution Ducts

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Shield Plugs

H-3-18679	Process Tube Outlet Shield Plug
H-3-18683	Process Tube Outlet Shield Plug
H-3-18684	Process Tube Outlet Shield Plug
H-3-18688	Process Tube Outlet Shield Plug

Moderator Storage Tank System

H-3-11566	Piping Moderator and Reflector Systems
H-3-11593	Piping Lower Reactor Room Arrangement
H-3-18670	Heavy Water Storage Tank Assembly
H-3-18671	Heavy Water Storage Tank Shell & Bottom Plate
H-3-18672	Heavy Water Storage Tank Top Plate
H-3-18673	Heavy Water Storage Tank Bottom Plate Detail & Assembly
H-3-18675	Heavy Water Storage Tank Top & Bottom Drain Details & Assy.

Primary Cooling System

H-3-11304	Reactor Outlet Plan
H-3-11305	Reactor Inlet Plan
H-3-11306	Reactor Inlet Sections
H-3-11360	Process Tube Flange Details
H-3-11577	(Page 1)Primary Coolant System Piping Arrangement
H-3-11577	(Page 2)Primary Coolant System Piping Arrangement
H-3-11577	(Page 3)Primary Coolant System Piping Arrangement

Biological Shield

H-3-11481	Biological Shield Plan Below EL -10'-0" Sections and Details
H-3-11482	Biological Shield Plan at EL 0'-0" Sections and Details
H-3-11484	Biological Shield Plans at -5'-0" and -20'-0" Moat Details
H-3-11517	Biological Shield Step Plugs

Thermal Barrier

H-3-11490	Thermal Barrier Arrangement Section
H-3-11507	Thermal Barrier Arrangement
H-3-11508	Thermal Barrier, Moveable Section, Top Plate
H-3-11510	Thermal Barrier Moveable Section Assembly
H-3-11511	Thermal Barrier Permanent Structure Plan & Details

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Rotating Shields

H-3-11540 Rotating Shield Arrangement

Miscellaneous Drawings

H-3-11007 Structural Containment Vessel Section and Details
H-3-11036 (Page 1) Structural Concrete Containment Vessel Plan at EL 0'-1' & Sections
H-3-11036 (Page 2) Structural Recesses for Moat Doors
H-3-11040 Structural Concrete Containment Vessel Plan at EL-11'-0" & Sections
H-3-11046 Steel Liner & Piping Lower Reactor Room
H-3-11059 Architectural General Floor Plan, 0'-0"
H-3-11060 Architectural General Floor Plan Below EL 0'-0"
H-3-11126 Service Piping Concrete Openings and Pipe Sleeves
H-3-11307 Process Tube Cross Section
H-3-11464 Reactor Gas Seal Outline & Drains
H-3-11592 Shim Control Tube D₂O Circulation Piping
H-3-11670 Process Cell Liquid Waste Collection System
H-3-11922 Instrumentation Steam Generation, Primary Moderator, Reflector, Purification & Power Calculating Systems - Equip & Turbine Arrangement in Cell "A"
H-3-14891 Piping & Inst. Engineering Flow Diagram, PRTR Decontamination, D₂O Cleanup
H-3-18639 Calandria Top & Bottom Primary Shields Mock-Up

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Meeting notes for PRTR Core Characterization meeting, 9/3/96

Discussed the various entry points, problems posed by each, and what information can be determined by each entry. The summary of conversation follows:

1. Make entry to the top face through the rotating shield step plug. This method has the advantage of being fairly straightforward. The process would be to chip out the grout around the plug and in the lifting bail socket, lift the plug with the crane, rotate it and pull it out.

Problems primarily consist of maintaining contamination control. This would most likely be done with a greenhouse and HEPA ventilation system. Other problems include chipping out the grout.

Potential information includes direct dose readings on the top face of the reactor, contamination smears of the top face, and possibly smears of inside one of the process tubes. Exposure rates would likely be rather low since the shield plug was designed to allow only narrow access to the top face.

2. Make entry to the side of the calandria through one of the instrument perforations through the biological shield. The process would be to re-certify the C-cell lifting points, fabricate a new step plug removal handle (H-3-11519) and lifting bolts (5/8" dia. rod, 5/8-11 UNC-2A x 4/3 LG threads [H-3-11517-1]), remove the step plug, and possibly thimble.

Problems include fabrication of equipment to enable plug removal, rigging problems associated with pulling out the plugs after so many years, and radiological concerns. If the thimble is also removed it is almost certainly contaminated on its outer surface.

Leaving the thimble in will allow a dose rate to be taken of the calandria forest through a relatively thin shielding of aluminum. This would be accomplished by inserting a "Tele" in the step plug socket until it touches the end of the thimble. Removing the thimble allows for more information by allowing access to the entire side of the calandria using borescope cameras. It would also allow for contamination information to be gathered directly off the calandria and would aid in the search for the 200 g of lost fuel from the September 1965 accident (although the fuel may have been burned up if it remained in the neutron flux, as it surely must by being inside the biological shield).

3. Make entry to the bottom face by entering the lower access space. This would be accomplished by activating the ventilation for the space, removing the "temporary" access plate, and making entry into the room.

Problems with this entry are almost entirely radiological. The lower access space was designed to be used up to very low power levels, so it has the ventilation system to control

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contamination. Mostly the problems are those typical of entering a known contamination area after it being isolated for nearly 30 years.

Types of information gleaned from this entry include contamination and dose level assessments, identifying which tube was pulled (in the 70's) to determine process tube dose rates, and some evaluation of the iron shot in the bottom shield. Additionally, this entry would facilitate a moderator dump tank inspection through the dump tank access plate.

4. The area beneath the moderator dump tank can be accessed through a 2'x2' drain hole from that area to the A-cell sump. This method was not seriously discussed as a primary access method into the reactor space since it offers so little information.

Questions raised in the meeting:

1. What made the "high density" concrete "high density?"
2. What are the other metals in the alloys labeled "mild steel" and "iron shot" and "cast iron?"
3. What are in the fission chambers and will we need to remove them?
4. What shape are the contents of the top and bottom shield - especially with not knowing whether the top shield was pumped out.
5. What are the contamination levels throughout the reactor core spaces?

PRTR Core Information Summary and Analysis

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APPENDIX B
PLUTONIUM RECYCLE TEST REACTOR LEAD SURVEY

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Table B-1. Building 309 Lead/Asbestos. (5 sheets)

Location	Component	Drawing/Zone	Mat'l/quantity
Containment	Fueling vehicle filter encloser cask	H-3-28013, sht 2	Lead/2,616 lb (Removed)
Containment	Profilometer unit cover plug assembly	SK-3-8155, sht 1/B12	Lead/1,355 lb
Containment C-Cell EI -1 ft 0 in.	Wire drive	II-3-14967, sht 1/C6 H-3-14967, sht 2/D4	Lead/66 lb (Removed)
Containment C-Cell	120KW electric water heater	H-3-11610/A3	SST and asbestos/ 8 cu in. (Removed)
Containment Approximately EI -24 ft 0 in. SE quadrant D2O recovery piping	Check valve (2)	H-3-11636/F8 H-3-11637	Asbestos/ 3/16 in. braided packing (Removed)
Containment Below EI -5 ft 0 in. A-Cell, NE quadrant	150# raised face flanges (Piping)	II-3-11652, sht 1	SST and asbestos/ ? All gaskets for 150# flanges (Removed)
Containment EI 10 ft 6 in. structural concrete cover slabs	Blow-out panels	H-3-11494, sht 1/E10	Lead/3,448 lb (Removed)
Containment EI 15 ft 6 in. structural concrete cover slabs	Lead bricks	II-3-11494, sht 1/A11	Lead/6,642 lb (Removed)
Containment EI 0 ft 0 in.	Water pit cover	II-3-11031/D6 H-3-11097	Lead/19,605 lb (Removed)
Containment EI -1 ft 0 in. SE quadrant steam drum enclosure	Steam piping relief line	H-3-11645/F2	Asbestos paper/ 488 cu in. (Removed)
Containment EI -21 ft 0 in. C-Cell	40,000# fuel element rupture detection system additional floor support	II-3-13685/E4	Lead/96 lb (Removed)
Containment EI -32 ft 0 in. A-Cell, SE quadrant	Helium storage tank supports	H-3-11501/C2	Asbestos/ 49 cu in. (Removed)
Containment EI -32 ft 0 in. NE quadrant reflector pumps	Pump anchor bolts	H-3-11566, sht 2/C1 II-3-11566, sht 1/A3	Lead/74 lb

Table B-1. Building 309 Lead/Asbestos. (5 sheets)

Location	Component	Drawing/Zone	Mat'l/quantity
Containment El -42 ft 0 in. NE quadrant moderator pumps	Pump anchor bolts	H-3-11566, sht 1/A3, D2	Lead/112 lb
Containment Secondary coolant activity monitor	Activity monitor	II-3-11820	Lead/ ? Lead shielding and lead plug (Removed)
Containment, center El -42 ft 0 in.	Heavy-water storage tank support	II-3-11502, sht 1/C3	Asbestos/ 206 cu in.
Containment use	Fuel handler cask and fuel handler skirt	II-3-14835	Lead/ ? (Removed)
Containment use	Gamma scanner	H-3-28062, sht 5/C11, 7/G4, and G8	Lead/1,008 lb (Removed)
Containment use	7 ton gatlign gun cask	H-3-20711, sht 1/G5, 2/G5, 3/F7, D11, and 4/C2	Lead/10,626 lb (Removed)
Fan pit	Exhaust air monitor chambers	H-3-15282/D6	Lead/36 lb
Fuel element rupture detection system	Gas sampler radiation plug	II-3-13703	Lead/11,815 lb (Removed)
Fuel element rupture facility	Pipe sleeve	SK-3-9238	Lead/2,021 lb Asbestos cement/916 cu in.
Fuel element rupture testing facility	Gaskets, parts 17, 20, 21, and 26	II-3-13394, sht 1/E3, B4, G9, H9	Stainless steel tank and asbestos/170 cu in.
Fuel element rupture testing facility	Gaskets, parts 19 and 23	II-3-13394, sht 1/G3, E9	Copper and asbestos/ 85 cu in.
Fuel examination facility	Floor plugs	H-3-11985/G9	Lead/1,106 lb
Fuel examination facility	1500 CFM air duct adapter	H-3-13372, sht 3/G10	Asbestos gasket/8 cu in.
Fuel examination facility	Corner shields	II-3-11987/D10	Lead wool/104 cu in.
Fuel examination facility	Air supply and exhaust system	II-3-11975/F3, G4 H-3-11990/H2	Asbestos cloth/35 cu in. (Removed)
Fuel examination facility	Wide angle viewer	H-3-9100, sht 1/B3, and 4/C7	Lead/239 lb
Fuel examination facility	250 CFM air duct adapter	H-3-13384/H7	Asbestos gasket/ 4 cu. in. (Removed)
Fuel examination facility	Service entry plug	H-3-11985/D3	Lead/672 lb

Table B-1. Building 309 Lead/Asbestos. (5 sheets)

Location	Component	Drawing/Zone	Mat'l/quantity
Fuel examination facility cover block EI 0 ft 0 in.	Viewing plug	H-3-11981/B6 H-3-11979	Lead glass/1,681 cu in.
Fuel examination facility cover block EI 0 ft 0 in.	Viewing plug	H-3-11981/B6 H-3-11979	Lead wool/45 cu in.
Fuel examination facility cover blocks	Cover block plugs	H-3-11978, sht 1/D4, and sht 2	Lead/2,681 lb
Fuel examination facility light fixture	Light fixture plugs	H-3-11982, sht 1/B2, and sht 2/D2	Lead/1,214 lb
Fuel examination facility wall block EI -26 ft 5 in.	Viewing plug	H-3-11981/H6 H-3-11979	Lead wool/41 cu in.
Fuel examination facility wall block EI -26 ft 5 in.	Viewing plug	H-3-11981/H6 H-3-11979	Lead glass/1,516 cu in.
Outside west of paint shop, brine pit	Loading hatch	H-3-11184, sht 3/F12	Lead/1,381 lb (Removed)
Outside containment east wall, EI 0 ft 0 in.	At 12 in. vent line	H-3-11494, sht 3/G7	Lead/1,785 lb (Removed)
Reactor structure	Sheet metal cone filled with lead	H-3-11550	Lead/2,252 lb
Reactor structure	PRTR shim control Mark III shim rod	H-3-13484, sht 2/C7 H-3-14913, sht 1/B7	Lead/287 lb
Reactor structure	Outer rotating shield disc	H-3-11541	Asbestos/ 646 cu in.
Reactor structure	Inner rotating shield	H-3-11550	Asbestos/ 194 cu in.
Reactor structure below bottom primary shield	Shim rod beam shield unit	H-3-11592/C4 H-3-15244/B8	Lead/136 lb
Reactor structure biological shield EI -2 ft 7 1/4 in. EI -17 ft 6 in.	D2O recovery system headers	H-3-11487	Asbestos/ 17,354 cu in.
Reactor structure bottom primary shield	Process tube inlet shield plug	H-3-11307/B9 H-3-11349	Lead/219 lb
Reactor structure lower reactor room	Steel liner and piping	H-3-11046/E8	Asbestos cloth/ 1,602 cu in.
Reactor structure moat openings	Moat stationary end shield	H-3-11443/B4 H-3-11444/B3, F3	Lead/3,328 lb
Reactor structure moat openings	Moat opening tip shielding	H-3-11484/C6	Lead/3,033 lb

Table B-1. Building 309 Lead/Asbestos. (5 sheets)

Location	Component	Drawing/Zone	Mat'l/quantity
Reactor structure moat openings	Moat doors	II-3-11444	Lead/30,230 lb
Reactor structure moat openings	Moat doors 1 and 4 gear plugs	H-3-11444/D10 H-3-11445/F10	Lead/418 lb
Reactor structure NW quadrant, EL -3 ft 6 in.	Biological shield penetration	II-3-11486, sht 2/A4, C5	Asbestos cement/ 1,032 cu. in. asbestos gasket/114 cu in.
Reactor structure NW quadrant EL -7 ft 10 in.	Lead shield	H-3-11482/B2 H-3-11484/G4	Lead/2,834 lb
Reactor structure process tubes	Process tube outlet nozzle assembly	II-3-11358/A3, Do	SST and asbestos/42 cu in.
Reactor structure stationary shield bottom assembly	Stationary shield	H-3-11535	Asbestos/957 cu in.
Reactor structure SW quadrant, EL -3 ft 6 in.	Access openings for future test loops	II-3-11482/C8 H-3-14967, sht 2/G9	Asbestos gasket/ 57 cu in. and asbestos/ 703 cu in.
Reactor structure thermal shield	Cooling loop tubes	H-3-11471/G5	Lead/2,657 lb
Reactor structure top primary shield	Ion chamber thimble	II-3-15281/B6, part 5	Lead/20 lb x number of part 5's (Removed)
Reactor structure top primary shield	Shim rod sander	H-3-15063, sht 2/D9	Lead/454 lb
Reactor structure top primary shield	Ion chamber thimble	II-3-15281/B2, part 1	Lead/20 lb x number of part 1's (Removed)
Reactor structure top primary shield	Center tube	H-3-15159, sht 1/A9, D11	Lead/117 lb
Reactor structure top primary shield	Moderator temperature probe	II-3-15090, sht 1/E5	Lead/21 lb
Reactor structure top primary shield	Moderator level indicator and temperature monitor	H-3-14449, sht 1/D5 and sht 2/B4	Lead/20 lb
Reactor structure top primary shield	Process shield tube outlet plug	II-3-11307/F2, H-3-18682 thru 84, and II-3-18679	Lead/4,156 lb
Reactor structure top primary shield	Ion chamber thimble	H-3-15281/B39, part 6	Lead/66 lb x No. of part 6's (Removed)

Table B-1. Building 309 Lead/Asbestos. (5 sheets)

Location	Component	Drawing/Zone	Mat'l/quantity
Reactor structure top primary shield	Ion chamber thimble	II-3-15281/G7, part 13	Lead/4 lb x No. of part 13's (Removed)
Reactor Structure top primary shield	Reflector vent line	H-3-11332/B3	Asbestos/1 cu in.
Reactor structure top primary shield	Moderator level probe	II-3-25885 sht 1/D2	Lead/21 lb
Shim rod transfer bucket	Storage can lid	H-3-13979/B5	Lead/11 lb x No. of can lids (Removed)
Storage basin	PRTR fuel handler filter box	H-3-13382, sht 2/B9, B10	Lead/684 lb (Removed)
Storage basin	Gamma scanner collimator shutter	H-3-15045, sht 3/G2, G8, G9, and E8	Lead/32 lb (Removed)
Total estimated lead			59,239 lb
Total estimated lead wool			190 cu in.
Total estimated lead glass			3,197 cu in.
Total estimated asbestos			24,087 cu in.

Cfm = Cubic foot per minute

NE = Northeast

NW = Northwest

PRTR = Plutonium Recycle Test Reactor

SE = Southeast

SST = Stainless steel tank

SW = Southwest

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APPENDIX C

Pictures of Characterization Work

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Table of Contents

Table Listing Archived Pictures of PRTR Characterization Activities	C-4
Removing Nozzle Cap	C-5
Removing Inner Rotating Shield	C-5
RO-7 Survey of Center Process Tube	C-6
Down-Well Gamma Counter Insertion	C-6
Removing Shield Plug 1L	C-7
Tools on Floor of Lower Reactor Space	C-7
Inside of Moderator Dump Tank, View 1	C-8
Inside of Moderator Dump Tank, View 2	C-8

This appendix contains pictures captured from video tapes of the work described in this document. Because they are originally video tape pictures they are not as clear as pictures taken on film. Table 1.0 lists reference numbers for pictures that will further illustrate Plutonium Recycle Test Reactor (PRTR) characterization work.

Table 1.0. Archived Pictures of PRTR Core Characterization

Reference number	Picture content
97020291-13CN	Top of reactor, center tube open
97020291-3CN	-21', thimble 1L with step plug removed
97020291-1CN	Thimble 1L dose survey with teletector
97020291-40CN	"Bull pen" setup around PRTR inner rotating shield
97020291-21CN	Under moderator dump tank through access hole
97020291-23CN	Access hole to space under moderator dump tank
97040252-20CN	Tools on floor of lower reactor space
97040252-43CN	Looking directly down on reactor top face, through outlet nozzles
97040252-60CN	Down thimble 1L, end of pipe is against reactor vessel wall
97040252-51CN	Lagging on outlet ring header, under stationary ring
97040252-66CN	Inserting down-well gamma probe into thimble 1L
97040252-58CN	Rigging in step plug 1L

Figure A. Removing Nozzle Cup

Figure B. Removing Inner Rotating Shield

Figure A. RO-7 Survey of Center Process Tube

Figure B. Down-Well Gamma Counter Insertion

Figure A. Removing Shield Plug IL

Figure B. Tools on Floor of Lower Reactor Space

Figure A. Inside of Moderator Dump Tank, View 1

Figure B. Inside of Moderator Dump Tank, View 2

APPENDIX D

WORK PLAN

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J-4A	Resolution/Retest (Continuation)		PTS Number 3B-96-00156
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1.0 SCOPE

Determine chemical and radiological status of the PRTR reactor core.

2.0 REFERENCES

- 2.1 H-3-11309 Side Flux Chamber Hole Assembly
- 2.2 H-3-11490 Thermal Barrier Arrangement & Section
- 2.3 H-3-11507 Thermal Barrier Arrangement
- 2.4 H-3-11518 Neutron Monitor & Fission Chamber Shielding Step Plugs
- 2.5 H-3-11519 Removal Handle, Step Plug
- 2.6 H-3-11540 Rotating Shield Arrangement
- 2.7 H-3-11852 -Instrumentation- Fission Chamber Control Arrangement & Details
- 2.8 H-3-11853 -Instrumentation- Fission Chamber Control Details
- 2.9 H-3-18640 Reactor Tube Index System
- 2.10 3B-96-00142, Grout Removal Work Package
- 2.11 Radiation Work Procedure (RWP) R-183
- 2.12 RWP R-184
- 2.13 RWP R-185
- 2.14 RWP R-186

3.0 MATERIALS AND EQUIPMENT

- 3.1 Portable Air Emission Unit (PTRAEU)
- 3.2 α and β Continuous Air Monitors (CAMs)
- 3.3 Rotating Shield Actuator, in working condition
- 3.4 Aluminized reinforced paper
- 3.5 Plastic sheeting
- 3.6 Plastic bags
- 3.7 4 large C-clamps to use for handles on temporary barrier.
- 3.8 2 Heavy Duty Carts
- 3.9 Cradles for shield plug and thimble
- 3.10 Personal Protective Equipment as specified in the RWPs
- 3.11 RCT to provide monitoring instruments and materials for smears
- 3.12 NDE personnel to provide all their equipment
- 3.13 Nuclear Chemistry personnel to provide all their equipment, as well as the RSR to ship their radioactive sources
- 3.14 Riggers to provide all lifting equipment they need for this job, excluding the crane.
- 3.15 Photography personnel to provide all their equipment
- 3.16 Packaging for transport and disposal of fission chambers

J-4A	Resolution/Retest (Continuation)	PTS Number 3B-96-00156
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- 3.17 Duct tape
- 3.18 Tape measure
- 3.19 Sturdy cord for lowering instruments into the process tubes
- 3.20 3/4" Allen wrench
- 3.21 3/8" Allen wrench
- 3.22 Special tool - nozzle cap remover
- 3.23 Wrenches
- 3.24 Large throat bolt cutters ($> 1/2"$) or hack saw
- 3.25 Bag for thimble (20" diameter x 10' long)
- 3.26 Temporary lighting
- 3.27 Waste containers
- 3.28 Fire Extinguisher, First Aid Kit, 2-way Radio Communications Equipment
- 3.29 Oxygen/Explosive Gas Meter

4.0 PRECAUTIONS AND LIMITATIONS

- 4.1 The top of the reactor has not been accessed for nearly 30 years. No historic radiological data exists. The reactor was used for several years after the 1965 accident so no Pu is expected here, but the area quite likely has high smearable contamination. This area needs to be entered cautiously and an RCT is required for the duration of this job.
- 4.2 Dose rates over the reactor core may be fairly high since there is not water in the process tubes. If dose rates are high enough to make the area on top of the reactor a High Radiation Area with the Inner Rotating Shield removed, the shield may be replaced when work is finished at the end of each day at the discretion of the RCT.
- 4.3 If, at any time, the RCT determines whole body dose rates or contamination levels exceed RWP Action Levels, core structures will be restored to "as found" condition and the need for the information re-evaluated against the expected exposure.
- 4.4 The Lower Reactor Space has not been accessed for nearly 30 years. No historic radiological data exists. The reactor was operated for several years after the 1965 accident so it is unlikely to be contaminated with Pu, however, expect a wide variety of fission products. This area needs to be entered cautiously. It is unclear from drawings whether the Thermal Barrier is resting on portable rests, the top of the Moderator Dump Tank, or has been removed. If the Thermal Barrier rests atop the Moderator Dump Tank video access through some avenue other than the access plate will be evaluated. Also, this area is a Non-permitted Confined Space.
- 4.5 A-cell sump is a Non-Permitted Confined Space, as is the space under the Moderator Dump Tank. Entry into these spaces must be done in accordance with appropriate safety precautions.
- 4.6 The A-cell sump is about 10' deep and has a permanent ladder. Evaluate the integrity of this ladder before use.

J-4A	Resolution/Retest (Continuation)		PTS Number 3B-96-00156
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- 4.7 Reactor Dome supply and HEPA filtered exhaust must be ON. When working on the 0' entry through the Inner Rotating Shield, recirculating heaters must be OFF.
- 4.8 In the event of material inadvertently being dropped down a process tube, it is possible to recover it from either the bottom of the tube directly or it can be pushed back up to the top of the tube. Either case will require evaluation of the importance of recovering the item and a separate work package. **DO NOT RECOVER IT IN THIS WORK PACKAGE.**
- 4.9 It may be necessary to relamp the -21' C-cell prior to performing work. If that is not possible, provide temporary lighting.
- 4.10 The trap door in the -21' C-cell floor has a load limit of 1,000 lbs. Limit the number of people on the door to a maximum of 4.

5.0 PREREQUISITES

- 5.1 Obtain NOC permit, RWPs, and gain approval of the Independent Review Group.
- 5.2 Confined Space Hazards Checklist complete.
- 5.3 Obtain all required tools, instruments, and materials.
- 5.4 Grout must be removed from the Inner Rotating Shield and the shield must be able to rotate (Work Package 3B-96-00142).
- 5.5 The "bull pen" must be erected around the Rotating Shields and include the Lay down Area.
- 5.6 Conduct a Pre-Job Briefing.
- 5.7 Coordinate sample transport to the 325 Laboratory.
- 5.8 Prior to conducting work in the -32' A-cell, ensure the RCT has completed the routine surveillance for that area.
- 5.9 Obtain work release signature from the Work Control Center.
- 5.10 Complete all prerequisites. _____ / _____

PIC Signature

Date

6.0 WORK INSTRUCTIONS

Note 1

Videotape (with audio) this work to document reactor conditions and to verify work completion.

- 6.1 **Access the Lower Reactor Space for Characterization** (Reference RWP R-185)
 - 6.1.1 Verify adequate ventilation in the space by using paper strips or ventilation smoke at the exhaust to show air flow.
 - 6.1.2 Set up CAMs as near as possible to the breathing zone.

J-4A	Resolution/Retest (Continuation)		PTS Number 3B-96-00156
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- 6.1.3 Remove retaining bolts from the northern temporary barrier to the lower reactor space in A-cell
- 6.1.4 Under constant RCT surveillance, remove the barrier and move it against the wall of the Fuel Examination Cell.
- 6.1.5 Establish temporary lighting in the lower reactor space
- 6.1.6 RCT perform radiological surveys of the lower reactor space (smears and dose rates), especially the following areas. Record readings for a Radiation Survey Report.
 - 6.1.6.1 Bottom of bottom shield
 - 6.1.6.2 Lower reactor space in general
 - 6.1.6.3 Piping systems that may be open (don't open any up for this purpose)
 - 6.1.6.4 Top of moderator dump tank

HOLD POINT: RCT perform radiological surveys.

Radiological conditions are below RWP Action Levels.

RCT Signature

Date

Note 2:

Steps 6.1.7 through 6.1.8 can be done in any order

- 6.1.7 Inspect the space for lead
 - 6.1.8 Videotape the moderator dump tank. Look for an easy access to the tank. If needed, videotape the tank using the following method:
 - 6.1.8.1 Remove access plate
 - 6.1.8.2 Video tape tank internals
 - 6.1.8.3 Replace access plate
 - 6.1.9 Replace the temporary barrier and properly post the area
- 6.2 Access the Space Under the Moderator Dump Tank** (Reference RWP-186)

J-4A	Resolution/Retest (Continuation)		PTS Number 3B-96-00156
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HOLD POINT: Test the air in the sump for adequate oxygen. Inspect the ladder for safety prior to using it.

The A-cell Sump has adequate oxygen and the ladder is safe for use.

PIC Signature

Date

HOLD POINT: Perform radiological characterization of A-cell sump. Document results on survey map.

Radiological conditions are below RWP Action Levels

RCT Signature

Date

- 6.2.1 Enter the A-cell sump and locate the access tunnel
 - 6.2.2 Set up temporary lighting in the access tunnel
 - 6.2.3 Perform radiological and video surveys of under the moderator dump tank
 - 6.2.4 Leave the A-cell sump and restore it to its "as found" condition
- 6.3 Access Top of Reactor for Characterization** (Reference RWP R-183)
- 6.3.1 Lift Inner Rotating Shield as follows:
 - 6.3.1.1 Establish the bull pen as a Contamination Area (CA).
 - 6.3.1.2 Ensure the Inner Rotating Shield is in the UNLOCKED position. If not, use the Rotating Shield Actuator to align the locking rings to the UNLOCKED position.
 - 6.3.1.3 Insert lifting bolts into the lifting lugs
 - 6.3.1.4 Set up the PTRAEU to ensure proper air flow as the Inner Rotating Shield is lifted.
 - 6.3.1.5 Lift the Inner Rotating Shield (6.8 tons) using the polar 30-ton crane (Lift shield in stages to allow for incremental survey as it is extracted)
 - 6.3.1.6 RCT perform radiological surveys as the Rotating Shield is removed.

J-4A	Resolution/Retest (Continuation)		PTS Number 3B-96-00156
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HOLD POINT: RCT perform radiological survey including dose rates and removable contamination.

Radiological conditions are below RWP Action Levels.

RCT Signature

Date

6.3.1.7 Place Inner Rotating Shield in designated lay down area and wrap it in plastic if directed by the RCT.

6.3.2 Perform Characterizations as follows:

- 6.3.2.1 Perform representative radiological survey (dose and contamination) of the accessible reactor top face. Indicate results on Data Sheet 1.
- 6.3.2.2 If directed by RCT, lay down plastic sheeting over the exposed reactor top.
- 6.3.2.3 Remove the top cap from up to four process tube nozzles (at the discretion of the project lead) and perform the following tests:

HOLD POINT: Survey the dose rate and contamination levels from the top of the nozzles with the top cap removed. Record results on Data Sheet 2.

Radiological conditions are below RWP Action Levels.

RCT Signature

Date

6.3.2.3.1 Test for water in the Top Shield by doing the following:

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- 6.3.2.3.1.1 Lower the PuBe/Cu water detection probe -30 inches into one of the process tubes (from the top of the nozzle) and secure (leave in place 12-24 hours at direction of Nuclear Chemist) (step 6.3.2.3.6 may be done while the probe is in place)
- 6.3.2.3.1.2 Remove the PuBe/Cu water detection probe from the process tube and test the Cu tube for neutron activation. RCT survey equipment as it is removed from the tube.
- 6.3.2.3.1.3 Place the PuBe source in a second Cu tube and lower it to -54". Secure it and leave it in place for the same length of time as previously used.
- 6.3.2.3.1.4 Remove the PuBe/Cu water detection probe from the process tube and test the Cu tube for neutron activation. RCT survey equipment as it is removed from the tube.
- 6.3.2.3.2 Characterize the reactor core by doing the following:
 - 6.3.2.3.2.1 Perform dose rate profile survey in three of the pressure tubes (record on Data Sheet 2). RCT survey equipment as it is removed from the tube.
 - 6.3.2.3.2.2 Insert the Gamma Energy Analysis probe and perform an isotopic profile in three of the process tubes (minimum of 5 elevations each). RCT survey equipment as it is removed from the tube.
 - 6.3.2.3.3 Replace all nozzle top caps (they can be left just hand tight).
- 6.3.3 Remove all test equipment from the top of the reactor.
- 6.3.4 Replace the Inner Rotating Shield
- 6.3.5 Survey and downgrade the bull pen area
- 6.3.6 Tape over the chipped paint areas.

6.4 Access the Side of the Reactor Core Through a Biological Shield Step Plug (Reference RWP R-184)

- 6.4.1 Establish a high radiation/contamination area around the shield step plug access and any areas that may become contaminated on the -21' level of C cell.
- 6.4.2 Set up CAMs near the breathing zone.
- 6.4.3 Stage waste (fission chambers) packaging in the room.
- 6.4.4 Set up the lifting cradle and position it at the lower step plug in -21' level of C-cell.
- 6.4.5 Remove all retaining nuts and the step plug cover plate.

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HOLD POINT: Survey under cover plate and on shield plug face.

Radiological conditions are below RWP Action Levels

RCT Signature

Date

- 6.4.6 Affix step plug removal hardware to the step plug

CAUTION:

Be prepared for moisture to come out of or be inside the thimble. This moisture will be D₂O.

- 6.4.7 Remove the step plug at a rate determined by the RCT to allow incremental survey of step plug
- 6.4.8 As the step plug is being removed, take radiological smears every 1' (use Data Sheet 3).
- 6.4.9 Cut off the fission chamber canister and package it for waste shipment.

HOLD POINT: Survey background and beam dose rates.

Radiological conditions are below RWP Action Levels.

RCT Signature

Date

- 6.4.10 Lower the step plug on to a hand cart and move it out of the way
- 6.4.11 Perform radiological surveys inside the thimble (smears and dose rates) (use data sheet 4)
- 6.4.12 Perform Gamma Energy Analysis along the thimble to the reactor vessel wall as a check of data gathered previously from the process tubes
- 6.4.13 Using the Gamma Energy Analysis instrumentation, check through the thimble wall for Pu.
- 6.4.14 Remove all test equipment from the thimble, RCT to survey equipment as it is removed.

J-4A	Resolution/Retest (Continuation)		PTS Number 3B-96-00156
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- 6.4.15 Replace the step plug, cover plate, and all retaining nuts and bolts. Nuts and bolts should be tight, but not torqued to specific levels.
- 6.4.16 RCT survey the area and downgrade the radiological controls

7.0 RECOVERY

Not applicable.

8.0 RETEST

Not applicable.

9.0 COMMENTS

[illegible]

J-4A	Resolution/Retest (Continuation)		PTS Number 3B-96-00156
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APPENDIX E

RADIO - CHEMISTRY REPORT ON WATER IN THE TOP SHIELD

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Battelle

Pacific Northwest Laboratories
Battelle Boulevard
P.O. Box 999
Richland, Washington 99352
Telephone (509) 376-0405

April 24, 1997

Bruce C. Cornwell
BWHC
Mail Stop L5-70
Richland, WA 99352

Dear Mr. Cornwell:

RESULT OF PNNL COPPER-ACTIVATION PROBE TO MEASURE MOISTURE IN PRTR REACTOR
(309 BUILDING) TOP THERMAL BARRIER

Reference: 1. Research Report for Westinghouse Hanford Company.
Reeder, P. L., R. L. Brodzinski, W. E. Wilson, D. C.
Stromswold, and J. H. Reeves. October 1995. "Moisture
Measurement for High-Level-Waste Tanks Using Copper
Activation Probe in Cone Penetrometer." PNL-10837. Pacific
Northwest Laboratory, Richland, Washington.

Enclosed is a set of data for work performed on various dates between **January 21, 1997**, and **February 27, 1997**, using the PNNL Copper-Activation Probe described in the referenced report to assess the presence of moisture in the PRTR Reactor (309 Building) top thermal barrier.

Neutron Techniques for Moisture Measurements

Neutron techniques have been used for measuring moisture in earth formations since the mid-1960s. Typically, a neutron source is lowered into an access well. Energetic neutrons from the neutron source enter the surrounding material and interact mainly with hydrogen atoms in any water that may be present. As the neutrons interact with water, they lose kinetic energy and become thermalized. Some of the neutrons with lower energy return to the instrument in the well and are available for detection. The closer the neutron detector is to the source of high-energy neutrons, the more sensitive the instrument is to surrounding moisture, i.e., a "near field" geometry.

The copper activation technique is insensitive to high gamma-ray backgrounds; in addition, only neutrons produce activation of copper. However, the activated copper must be removed from the well in order to measure its activation. Thus, the results are not available in "real-time" but are delayed by several hours.

B. C. Cornwell
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PNNL Copper-Activation Probe for Moisture Measurements

In the PNNL copper-activation method, a source of high-energy neutrons provides the means for activation; the particular source used in these sets of experiments was $^{239}\text{PuBe}$ source M-710 containing 1.2 g of Pu and emitting $1.58 \cdot 10^5$ n/s. The average neutron energy is about 4.5 MeV with a maximum energy of about 10.7 MeV. Thermal neutrons at room temperature are usually defined as having a kinetic energy of 0.025 eV.

Figure 1 shows a schematic diagram of the assemblage used to measure moisture in the PRTR thermal barriers. The high-energy-neutron source is secured inside a hollow copper pipe using an aluminum holder that maintains the source at the center of the copper tube. The neutron-source/copper-tube assemblage was suspended in one of the process tubes of the reactor. Fast neutrons from the source readily penetrate the copper and steel to enter the material in the thermal barrier. Collisions of the neutrons with any hydrogen atoms provide the most efficient means for slowing down the neutrons since the low mass of hydrogen atoms is essentially the same as that of the neutron and allows efficient energy transfer. Iron in the thermal barrier also slows down the neutron, but with about an efficiency of 6% relative to 100% for hydrogen. After the neutrons have lost their energy, they continue to scatter, gaining and losing some energy in the thermal energy region. Some of these thermal neutrons return to the probe where they have an opportunity to

Figure 1 Schematic Diagram of the PNNL Copper Probe Assemblage for Measuring Moisture in the PRTR Thermal Barriers.

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be absorbed to activate the copper. The activity of the copper continues to build up from exposure to the neutrons until a state of equilibrium develops in which the rate of production of newly activated copper nuclei is exactly equal to the rate of decay of previously activated ones. Activation that lasts for two half lives will produce 75% of the equilibrium amount; in practice, it is not worth prolonging the activation because of time constraints and because sufficient activation has usually been achieved after 1 or 2 half lives.

After the high-energy neutrons slow down to thermal energy through collisions with surrounding material, they can be absorbed by ^{63}Cu (natural abundance of 69.2%) to become ^{64}Cu , which decays with a half life of 12.7 h. The mode of decay which yields a readily detectable signal is via positron emission:

The emitted positron (β^+) annihilates with an electron to give two gamma rays, each with energy 0.511 MeV, which are emitted simultaneously in opposite directions. Two gamma-ray detectors, located on opposite sides of the source of the positrons, detect the emitted gamma rays. Requiring the gamma rays to have the proper energy and also be detected in coincidence greatly reduces the background counts. Greater sensitivity is achieved by further lowering the background with an anti-coincidence detector (anti-cosmic or veto shield) and massive external lead shielding.

PNNL Multi-Dimensional Gamma-Ray Counting System

Appendix A contains a figure that illustrates one of the features of the multi-dimensional gamma-ray counter used to measure the activation of the copper pipes. The two large NaI(Tl) scintillation detectors each have a diameter of 30 cm and a thickness of 20 cm. Each NaI(Tl) detector is surrounded by a plastic anti-coincidence detector to reduce background. Individual copper pipes were placed on their side between the NaI(Tl) detectors and inside concentric annuluses of lead and gold to provide the most efficient counting geometry. The annuluses reduce Compton backscatter between the two NaI(Tl) detectors and increase the likelihood of positron annihilation within the sample layer (i.e., the volume nearest both detectors.)

The computer recording data pulses identifies both single- and coincident events by polling two analog-to-digital converters (ADCs) that convert detected scintillation pulses into digital values. Data were only accepted by the electronics for processing when there was no simultaneous pulse from the anti-coincidence shield, which typically serves as a signal that a muon (or some other

B. C. Cornwell
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cosmic ray) has traversed the detector volumes and will merely elevate background values. Valid data registering sufficiently close together were deemed to be coincident; all other valid data was treated as single events. The collection of pulse-height data were stored as data in a two-dimensional array of 64 channels on each axis, covering the energy range from 0.1 to 3 MeV. Counting times typically summed to more than a day of total counting.

The counts in five channels of the two-dimensional data array that corresponded to the 0.511-0.511 keV coincidence peak were summed to give the ^{64}Cu count rate. Appendix B contains a grid that displays some of the radionuclides that have been identified and quantified using the multi-dimensional gamma-ray spectrometer. Positron annihilation is shown in the coincidence plane of the grid at the intersection of channels (11,11) as a purple square. Note that regions identified in the coincidence plane are mirrored across a diagonal that extends between the coordinates (0,0) and (63,63) corresponding to detection in the upper vs. the lower NaI(Tl) detector.

The data analysis program MITE2 was used to sum the specified channels, subtract detector background, estimate contributions from other Compton tails, and correct the count rate back to the time the copper pipe was removed from the reactor process tube.

Data Analysis and Results

Table 1 summarizes the MITE2 results from using the PNNL Copper Probe to measure moisture content. The first entry shows the results for a copper pipe counted in the multi-dimensional gamma-ray detector without exposing the pipe to the high-energy neutron source; the result, expressed as a percentage relative to exposure to water, shows a very low background. The second entry shows the probe immersed in a large volume of water to provide a comparison for the upper end of the measurement. The third and fourth entries show the results of the count for probes placed at the top and bottom portions of the upper PRTR thermal barrier. The fact that their water equivalence is nearly identical indicates that if there were water present, it would be at the same concentration throughout the thermal barrier and would not be pooled in the bottom layer of the thermal barrier. The last measurement was performed on the lower PRTR thermal barrier, which was known to be drained of all water; the measurement was taken in the middle of the thermal barrier, so that a greater fraction of the neutron thermalized by the surrounding iron were reflected back toward the copper and fewer escaped into the surrounding void. The last value also confirms that the upper PRTR thermal barrier has been drained of water and that any remaining hydrogen is at a very low level.

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Table 1. Summary of MITE2 results from using the PNNL Copper Probe to measure moisture content in the PRTR thermal barriers.

Location of Cu Probe	Sample ID	Irradiation Time (min.)	Water Equivalent
Unexposed Copper Pipe	F007	--	0.00063%
Tank of Water	F007	1522	100.0%
Top of Upper PRTR Thermal Barrier	D007	1362	1.75%
Bottom of Upper PRTR Thermal Barrier	E001	1397	1.86%
Middle of Lower PRTR Thermal Barrier	G003	1383	2.15%

Recommendations and Conclusions

Copies of the data used to confirm that water has been drained from the PRTR thermal barriers are maintained in the PNNL NDA System project file. Spectral data are archived on computer media for backup purposes. Any questions may be directed to me at Mail Stop P8-08, by cc:Mail, by FAX at 376-2329, or by phone at 376-0405.

Sincerely,

Richard J. Arthur
Sr. Research Scientist

Enclosures
Attachments

cc w/attachments:
NS Hale, L5-70
MD Winterrose, P8-01
File /LB

Appendix A

Features of the PNNL Multi-Dimensional Gamma-Ray Counter
Used to Measure Low-Level Activation in Copper Pipes

Appendix B

A Grid That Displays Radionuclides Measured With
the PNNL Multi-Dimensional Gamma-Ray Spectrometer

Multi-Dimensional Gamma-Ray Spectrometer

Appendix C

Diagram Illustrating Sampling Locations, Marked as "**", Within the PRTR
Using the PNNL Moisture-Measuring Copper Probe
Used to Measure Low-Level Activation in Copper Pipes

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APPENDIX F

Radiological Survey Report

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Westinghouse Hanford Company - Radiological Control RADIOLOGICAL SURVEY REPORT				Survey No. R 000051						
Date 01/13/1987	Time 0900-1400	RWP No.(s) R-003 REV 006	Page <u>1</u> of <u>5</u>							
Area/Bldg./Room/Location (Code) <u>300 AREA / 209 BLDG / -32' LEVEL CONTAINMENT T</u>					F.C. <u>R</u>					
Job Description <u>PERFORM UPON ENTRY ROUTINE</u> <u>WFOO1 ON -32' LEVEL OF CONTAINMENT</u> <u>PRIOR TO UPCOMING WORK.</u> <u>1/1</u>			Purpose of Survey (check appropriate boxes): Contamination Incident: ☐ Skin, ☐ Clothing, ☐ Spill Alarm Response: ☐ CAM, ☐ ARM(RAM), ☐ PSD ☐ HRA/VHRA Work ☐ Job Coverage ☐ Other ☐ Exposure Incident ☐ Material Release ☐ RM Transfer/Shipment ☒ Required, Task No. <u>WFOO1</u>							
<div style="text-align: right; margin-bottom: 10px;">Map/Sketch</div> <div style="text-align: center; font-size: 2em; transform: rotate(-30deg); opacity: 0.5;">SEE ATTACHED MAPS</div>										
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS				
			μ (non-pen) mrad/h	γ (pen) mR/h	α mrem/h	Total (1/100 cm ²)		Removable (1/100 cm ²)		
			μ (dpm)	α (dpm)	β (dpm)	γ (dpm)	α (dpm)	β (dpm)	γ (dpm)	mrad/h
1	BUFFERS AREA (MAX)	1/1				8000	<D	<D	<D	<D
2	STEP OFF PAD	2/1				<D	<D	<D	<D	1/1
3	"B" CELL (MAX)	1/1				10000	<D	10000	<D	1/1
4	"C" CELL (MAX)	C	8	1	1/1	<D	5000	140	<D	1/1
5	"A" CELL (MAX WALL)	C	20	4	1/1		20000	<D	<D	1/1
6	"A" CELL (MAX FLOOR)	2/1					2000	280	<D	1/1
7	SUMP AREA FLOOR	1/1					2000	140	<D	1/1
8	TUNNEL (SUMP TO BUFFERS) LEVEL	1/1					50000	7000	<D	1/1
9	SEE ATTACHED MAPS FOR MORE INFO	1/1								?
10										

Continued on page 2

Air Sample Results (μ Ci/ml)				Legend	
BZ	GA	Initial	Decay	①	△
21	1/1			① - Smear Location	△ - Air Sample Location
21				② - Large Area Smear	• - Contact Reading
22				Other <u>1/1</u>	
22					

BD-6000-010 106/95

WHC - Radiological Control RADIOLOGICAL SURVEY REPORT				F.C.	Page <u>2</u> of <u>5</u>	Survey No. <u>R000051</u>				
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS				
			■ (non-pen) mrad/h	γ (pen) mR/h	■ mrem/h	Total (100 cm ²)		Removable (100cm ²)		
						■ (dpm)	■ (dpm)	■ (dpm)	■ (dpm)	mrad/h
1A										
<div style="position: absolute; top: 0; left: 0; width: 100%; height: 100%; background: linear-gradient(to top right, transparent 49%, black 49%, black 51%, transparent 51%); background-size: 100% 100%; pointer-events: none;"></div>										
Comments: <u>Total <D = 2 inches / sec</u> <u>Removal <D = 1 inches / sec</u> <u>0.25 inches from item</u> <u>0.25 inches from item</u> <u>10 % of item surveyed</u> <u>100 % of item surveyed</u> <u>No audible counts above background</u> <u>No audible counts above background</u> * NO CORRECTION FACTORS APPLIED FOR SOURCE SIZE OR TOTAL READINGS * REMOVABLE AND DIRECT ARE NOT ADDED TOGETHER IN TOTAL COLUMN * ALL SURFACES WERE LARGE AREA WIPES. * SWAB IN "A" CELL SURVEYED USING REACH ROD, NO ACCESS MADE * SEE MAPS FOR MORE INFORMATION AND DETAILS. <u>1A</u>										
Samples Counted with portable instruments for <u>1A</u> minutes.										
Instr./Probe Model	RO-3B (CP)	E-140/Pencak (GM/Probe)	Surveyor-X w/ZnS probe (PAMI)	1A						
Serial No.	1CE83-0419	CME87-00A7 DTE89-00A7	ACW22-00A7							
Efficiency	100%	10%	14%							
Correction Factor	1	10	7							
ADDITIONAL REPORTS										
RPR No. _____ Sample Counter Log(s) _____ RSR No. <u>1A</u> ... Contamination - Skin _____ Procedure: <u>1A</u> ... Contamination - Clothing <u>1A</u> Other(s) _____										
HPT: Payroll No.: <u>95337</u> Date: <u>1-14-97</u> Payroll No.: <u>95641</u> Name (Print): <u>C. WILMAN</u> Name (Print): <u>G.A. DAVIS</u> Signature: <u>[Signature]</u> Signature: <u>[Signature]</u>

309 BLDG. CELL "B" -32' LEVEL

309B10.BKG 01-10-97 pw

- Fixed Contamination
- ★ Removable Contamination

HPT GAD, Date 1/13/97

SURVEY # R000051 PAGE 4 OF 5

Westinghouse Hanford Company - Radiological Control RADIOLOGICAL SURVEY REPORT					Survey No. R000052					
Date 01/15/1997		Time 0800-1300		RWP No.(s) R-185		Page 1 of 45				
Area/Bldg./Room/Location (Code) 300 AREA 509 BLDG 32' LEVEL CONTAINMENT WUDGE REACTOR CORE							F.C. R			
Job Description PERFORM CHARACTERIZATION OF AREA WUDGE REACTOR CORE AFTER REMOVING BARRIER WALL				Purpose of Survey (check appropriate box(es)): Contamination Incident: ☐ Skin, ☐ Clothing, ☐ Spill Alarm Response: ☐ CAM, ☐ ARM(RAM), ☐ PSD ☐ HRA/VHRA Work ☐ Job Coverage ☒ Other ☐ Exposure Incident ☐ Material Release ☐ RM Transfer/Shipment ☐ Required, Task No.						
Map/Sketch SEE ATTACHED MAPS										
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS				
			h (non-pen) mrad/h	g (pen) mR/h	g mrem/h	Total (/100 cm ²)		Removable (/100cm ²)		
			g (dpm)	g (dpm)	g (dpm)	g (dpm)	g (dpm)	g (dpm)	g (dpm)	mrad/h
1	INTERIOR OF BARRICADE	F	1/2	2	1/2			4000	350	1/2
2	FLOOR (MAX)	C	1/2	10	1/2			50000	350	1/2
3	WALL (MAX)	1/2						10000	700	1/2
4	CEILING (MAX)	C	1/2	8	1/2			7000	1400	1/2
5	DUMP TANK PIPES (MAX)	1/2						130000	2100	1/2
6	WALKWAY (MAX)	F	1/2	3.5	1/2			70,000	310	1/2
7	GENERAL ROOM DOSE RATE	1/2								
	AT CEILING (MAX)	15cm	1/2	5	1/2					
8	DOSE RATE AT FOOT LEVEL (MAX)	15cm	1/2	5	1/2					
9	GENERAL ROOM	F	1/2			40.1	1/2			
Continued on page 2										
Air Sample Results (uCi/ml)					Legend					
BZ	CA	Initial	Decay		☐ - Smear Location	☒ - Air Sample Location				
#1	A	1/2	6.6 x 10 ⁻¹³		☐ - Large Area Smear	* - Contact Reading				
#1	X	1/2	4.17 x 10 ⁻¹²		Other 1/2					
#2	1/2									
#2										

BD-6000-010 (06/95)

WHC - Radiological Control RADIOLOGICAL SURVEY REPORT				F.C.	Page	Survey No.					
				2	2 of 4	R000052					
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS					
			h (non-pen) mrad/h	γ (pen) mR/h	n mrem/h	Total (1/100 cm ²)		Removable (1/100cm ²)			
						μldpm	μldpm	μldpm	μldpm	μldpm	mrad/h
10	BIO SHIELD JACK TABLE (MAX)	2 1/4						100000	10300		2 1/4
11	RAILING (MAX)	2 1/4						23000	320		2 1/4
12	STEP OFF PAD	2 1/4						11000	220		2 1/4
		2 1/4									
/											
Comments											
* ALL SURFACES WERE CAREFULLY WIPED TO CHARACTERIZE THE ROOM.											
* DOSE RATE READINGS SHOW HIGHEST READINGS AT FLOOR OR CEILING WITH WHOLE BODY DOSE BETWEEN TWO READINGS.											
* SEE ATTACHED MAPS FOR MORE INFORMATION.											
* SURVEY WILL CONTINUE TO BETTER CHARACTERIZE ROOM TOMORROW.											
* THERE IS A HIGH DUST LEVEL IN ROOM WITH MISC. TOOLS SCATTERED ON FLOOR.											
2 1/4											
/											
Samples Counted with portable instruments for 2 minutes.											
Instr./Probe Model	RO-3B (CP)	E-140/Pancake (GM/Probe)	Surveyor-X w/ZnS probe (PAM)	30004	1						
Serial No.	1CEB3-0149	2MEBS-0094 DTES-0097	ACHN2-0047	3044	1						
Efficiency	100%	10%	14%	100%							
Correction Factor	1	10	7	1							
ADDITIONAL REPORTS											
APR No.	1	Sample Counter Log(s)	R-01151597-219	R-01151597-219 (A/C/E/R/A/N/D)							
RSR No.	1	Contamination - Skin		Procedure:	2 1/4						
		Contamination - Clothing	2 1/4	Other(s)	1						
HPT:	Payroll No.: 95837	First-Line HP Manager:	Date: 11/7/97	Payroll No.: 64783							
Name (Print): C. WILDMAN	Name (Print): D. R. ELLISON										
Signature: C. Wildman	Signature: D. R. Ellison										

BD-6000-010A (06/96)

Westinghouse Hanford Company - Radiological Control RADIOLOGICAL SURVEY REPORT				Survey No. R 000053						
Date <u>01/16/1997</u>		Time <u>0800-1300</u>		RWP No.(s) <u>R-185</u>						
Area/Bldg./Room/Location (Code) <u>200 AREA 309 Bldg. -32' CONTAINMENT / W/DRY REACTOR CORE</u>				Page <u>1</u> of <u>24</u>						
Job Description <u>PERFORM AREA MONITORING DUMP</u> <u>TANK THEN ACCESS HATCH ON TANK</u> <u>TOP, AND TECH SMears OF WORK AREA</u>				F.C. <u>2</u>						
Purpose of Survey (check appropriate box(es)): Contamination Incident: ☐ Skin, ☐ Clothing, ☐ Spill Alarm Response: ☐ CAM, ☐ ARMIRAM, ☐ PSD ☐ HRA/VHRA Work ☐ Job Coverage ☒ Other ☐ Exposure Incident ☐ Material Release ☐ RM Transfer/Shipments ☐ Required, Task No. _____										
Map/Sketch										
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS				
			h (non-pen) mrad/h	g (pen) mR/h	g mrem/h	Total (/100 cm ²)		Removable (/100cm ²)		
			g (dpm)	g (dpm)	g (dpm)	g (dpm)	g (dpm)	g (dpm)	g (dpm)	mrad/h
1	HATCH	C	1/4	10	1/4			20,000	1,000	1/4
2	OPEN TANK AT TOP	C	1/4	10	1/4					
3	TANK MAX DOSE RATE	F	1/4	70	1/4					
4	SMEAR OF TANK INTERIOR	C	1/4	1500	1/4					15
5	TECH SMEAR OF TANK INTERIOR	1/4						1500	100	1/4
6	TANK CLOSED	C	1/4	10	1/4			10,000	400	1/4
7	REMOVABLE WALL REPLACES	C	1/4	2	1/4			2000	40	1/4
8	FLOOR BY WALL	1/4						4000	40	1/4
Continued on page 2										
Air Sample Results (uCi/ml)						Legend				
	BZ	GA	Initial	Decay		☒ - Smear Location	☒ - Air Sample Location			
#1	X		1/4	2.07 x 10 ⁻¹²		☒ - Large Area Smear	* - Contact Reading			
#1	Y		1/4	4.28 x 10 ⁻¹²		Other <u>1/4</u>				
#2	1/4									
#2										

SD-8000-010 (06/96)

BD-5000-010R (06/95)

Page	
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B-01171997-219

1 of 1

[illegible]

For example, using a 50- μ m opening filter, Halloworth & Voss HV-70, 113-5211, or Gelman Versapore AN-30001, the fractional filter collection efficiency $E_F = 95/100 = 0.95$.
 The flow rate used in the following table units is l/min, cfm, scfm, or gpm.

Definitions:

ϕ_{air} = Counting efficiency (counts/mL); CF = Correction factor ($1/\text{CF}_{\text{air}}$); N_0 = Number of background counts observed; T_B = Background counting time (min); N_B = Background count rate (cpm); N_1 , T_1 , N_2 , T_2 , N_3 = General Area, BZ = Breathing Zone; T_C = Time sample counting started (124 hr clock); T_D = Gross sample counting time (min); MDC = Minimum detectable concentration ($\mu\text{Ci}/\text{mL}$); N_G = Number of gross sample counts observed; N_N = Net sample count rate (cpm); N_{air} , T_{air} = R-15; C = Concentration ($\mu\text{Ci}/\text{mL}$); $\text{CF} = 0.95$; Filter media collection efficiency; V_A = Volume of air sampled; $k = 6.736 \times 10^{-3} \text{ h V}_A$ in h^2 , or $k = 2.22 \times 10^{-3} \text{ h V}_A$ in liters or $k = 7.22 \times 10^{-3} \text{ h V}_A$ in m^3 ; λ in m^3 ; $\lambda = 0.5$ (volume) and 0.95 (beta) for impinger/planchet's self absorption factor.

Log Validated By (Print Name and Sign): _____

G. A. DAVIS

Date. 1-17-97

A-6000-274 (11/96)

Westinghouse Hanford Survey - Radiological Control RADIOLOGICAL SURVEY REPORT						Survey No. R 000054				
Date 01/16/1997		Time 1300-1500		RWP No.[s] R-186		Page 1 of 3				
Area/Bldg./Room/Location (Code) 300 AREA / 300 BLDG / -3D LEVEL CONTAINMENT / "A" CELL SUMP.						F.C. R				
Job Description <u>PURPOSE CHARACTERIZATION OF</u> <u>AREA UNDER MODERATE DUMP TANK</u> <u>ACCESSED FROM THE "A" CELL SUMP.</u>				Purpose of Survey (check appropriate box(es)): Contamination Incident: ☐ Skin, ☐ Clothing, ☐ Spill Alarm Response: ☐ CAM, ☐ ARM(RAM), ☐ PSD ☐ HRA/VHRA Work ☐ Job Coverage ☒ Other ☐ Exposure Incident ☐ Material Release ☐ RM Transfer/Shipment ☐ Required, Task No.						
Map/Sketch 										
DOSE RATES CONTAMINATION LEVELS										
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS				
			■ (non-pen) mrad/h	γ (pen) mR/h	■ mrem/h	Total (/100 cm ²)		Removable (/100cm ²)		
						■ (dpm)	■ (dpm)	■ (dpm)	■ (dpm)	mrad/h
1	SUMP FLOOR	F	<1	<1	<1			2000	140	<1
2	LEDGE	>1						5000	7000	<1
3	OPENING TO TANK	F	25	3	>1					
4	GENERAL AREA BY OPENING	F	2	1	>1			2000	140	<1
5	TENH SUMP FLOOR UNDER TANK	>1						4000	140	<1
6	TENH SUMP TANK	>1						4000	420	<1
Continued on page 2										
Air Sample Results (uCi/ml)						Legend				
BZ	GA	Initial		Decay		(#) - Smear Location	(▲) - Air Sample Location			
#1	X	<1		2.07 x 10 ⁻²		(#) - Large Area Smear	* - Contact Reading			
#2	X	<1		4.28 x 10 ⁻²						
#3										
#4										
#5										
#6										
Other _____										

WHC - Radiological Control RADIOLOGICAL SURVEY REPORT				F.C.	Page <u>2</u> of <u>3</u>	Survey No. R000054				
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS				
			μ (non pen) mrad/h	γ (pen) mR/h	μ mrem/h	Total (100 cm ²)		Removable (100cm ²)		
						μ (dpm)	μ (dpm)	μ (dpm)	μ (dpm)	mrad/h
1										
Comments <u>* SEE ATTACHMENT "A" FOR FURTHER DETAILS OF TANK SURVEY.</u> <u>* REFER TO SURVEYS R000052, R000053 FOR MORE INFORMATION</u> <u>* SAMPLES WERE TAKEN AND WILL BE SENT TO THE LBS FOR ANALYSIS</u> <u>4h</u>										
Samples Counted with portable instruments for <u>0.25</u> minutes.										
Instr./Probe Model	RO-3B (CP)	E-140/Pancake (GM/Probe)	Surveyor-X w/ZnS probe (PAM)	R-07	4h					
Serial No.	16E83-0189	CNE65-0094 BTE69-0057	ACM02-0047	16E82-0036						
Efficiency	100%	10%	14%	100%						
Correction Factor	1	10	7	1						
ADDITIONAL REPORTS										
APR No.	Sample Counter Log(s)		R-0117197-119							
RSR No.	Contamination - Skin		Procedure: 4h							
Contamination - Clothing		4h Other(s)								
HPT: Payroll No.: <u>9837</u>										
First-Line HP Manager: Date: <u>11/7/97</u> Payroll No.: <u>6A783</u>										
Name (Print): <u>C. W. Johnson</u> Name (Print): <u>D. R. Ekstrom</u>										
Signature: <u>C. W. Johnson</u> Signature: <u>D. R. Ekstrom</u>										

BD-6000-010R (06/96)

ATTACHMENT "A"

SURVEY OF MODERATOR DUMP TANK

USING RO7 FOR SURVEY

HATCH OPEN	10 MR/HR	0' LEVEL (TOP OF TANK @ OPENING)
TANK INTERIOR	10 MR/HR	1' DOWN
	10 MR/HR	2'
	10 MR/HR	3'
	20 MR/HR	4'
	40 MR/HR	5'
	70 MR/HR	5.5' ---BOTTOM

ACCESS TUNNEL UNDER TANK

USING RO7 FOR SURVEY

0 MR/HR	OPENING (SEE NOTE 1)
0 MR/HR	1' IN FROM SUMP TO TANK CENTER
0 MR/HR	2'
0 MR/HR	3'
10 MR/HR	4'
0 MR/HR	5'
0 MR/HR	6'
10 MR/HR	7'
0 MR/HR	8'
10 MR/HR	9' UNDER CENTER OF TANK SUMP

(READINGS ARE AT THE BOTTOM SCALE OF THE INSTRUMENT)

NOTE 1: READINGS WITH CP
 IN "A" CELL SUMP AT OPENING TO DUMP TANK
 AT OPENING 25 MR/HR W/O, 3 MR/HR W/C
 GENERAL AREA 2 MR/HR W/O, 1 MR/HR W/C

REFERENCE SURVEYS R000052, R000053, R000054 FOR COMPLETE INFORMATION ON TANK SURVEY.

SURVEY # R000054 PAGE 3 OF 3
 HPT: C. WILDMAN 01/16/1997

Westinghouse Hanford Company - Radiological Control RADIOLOGICAL SURVEY REPORT					Survey No. R 000058					
Date 1/17/1997		Time 0800-1200		RWP No.(s) R-003 REV006		Page <u>1</u> of <u>2</u>				
Area/Bldg./Room/Location (Code) 300 AREA / 109 Bldg. / -32' LEVEL CONTAINMENT						F.C. A				
Job Description REMOVE ALL TOOLS AND EQUIPMENT USED ON JOB ACCESSING UNDER REACTOR CORE.				Purpose of Survey (check appropriate boxes): Contamination Incident: ☐ Skin, ☐ Clothing, ☐ Spill Alarm Response: ☐ CAM, ☐ ARM(RAM), ☐ PSD ☐ HRA/VHRA Work ☒ Job Coverage ☐ Other ☐ Exposure Incident ☐ Material Release ☐ RM Transfer/Shipment ☐ Required, Task No.						
Map/Sketch 										
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS				
			α (non-pen) mrad/h	γ (pen) mR/h	n mrem/h	Total (1/100 cm ²)		Removable (1/100cm ²)		
			α (dpm)	β (dpm)	γ (dpm)	α (dpm)	β (dpm)	γ (dpm)	α (dpm)	mrad/h
1	CAMERAS & EQUIPMENT	N/A	<D	<D	<D	<D	<D	<D	<D	N/A
2	LIGHTS	N/A	<D	<D	<D	<D	<D	<D	<D	N/A
3	CORDS / CABLES	N/A	<D	<D	<D	<D	<D	<D	<D	N/A
4	CAMS & COOLWATER	N/A	<D	<D	<D	<D	<D	<D	<D	N/A
5	MISC EQUIPMENT	N/A	<D	<D	<D	<D	<D	<D	<D	N/A
Continued on page 2										
Air Sample Results (μCi/ml)					Legend					
BZ	GA	Initial	Decay		(P)	△				
#1	N/A				(P) - Smear Location	△ - Air Sample Location				
#1					(F) - Large Area Smear	• - Contact Reading				
#2					Other <u>N/A</u>					
#2										

BD-6000-010 (05/86)

Westinghouse Hanford Company - Radiological Control RADIOLOGICAL SURVEY REPORT						Survey No. R 000060				
Date 01/21/1997		Time 0800-1500		RWP No.(s) R-183		Page <u>1</u> of <u>23</u>				
Area/Bldg./Room/Location (Code) 300 AREA / 309 BLDG / CONTAINMENT D'LEVEL						F.C. R				
Job Description REMOVE ROTATING SHIELD FROM OVER REACTOR CORE AND PERFORM CHARACTERIZATION OF PROCESS TUBES WITHIN SHIELD ACCESS.				Purpose of Survey (check appropriate boxes):						
				Contamination Incident: ☐ Skin, ☐ Clothing, ☐ Spill Alarm Response: ☐ CAM, ☐ ARM(RAM), ☐ PSD ☐ HRA/VHRA Work ☐ Job Coverage ☒ Other ☐ Exposure Incident ☐ Material Release ☐ RM Transfer/Shipments ☐ Required, Task No.						
Map/Sketch 										
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS				
			$\dot{H}$ (non-pen) mrad/h	$\dot{H}$ (pen) mR/h	$\dot{H}$ mrem/h	Total (/100 cm ²)		Removable (/100cm ²)		
			$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	$\dot{H}$ (dpm)
1	ROTATING SHIELD (MAX)	1/2						1000	140	1/2
2	FLOOR LEVEL	F	1/2	0.8	1/2					
3	PORT WALLS (MAX)	1/2						2000	280	1/2
4	PROCESS TUBE LEVEL (GENERAL)	F	2.5	3	1/2					
5	TUBE 1247 (MAX D/2)	C	14	2.5	1/2					
6	TUBE CAPS (GENERAL)	6 CM	2.5	3	1/2			2000	420	1/2
7	FLOOR GENERAL (MAX)	1/2						6000	700	1/2
8	FLOOR BETWEEN TUBES (MAX)	1/2						20000	2100	1/2
9	DOSE BETWEEN TUBES WITH CAP (MAX)	F	1/2	5	1/2					
10	TUBE 1049 CAP REMOVED	C	1/2	2	1/2			10000	140	1/2
Continued on page 2										
Air Sample Results (μ Ci/ml)					Legend					
BZ	GA	Initial	Decay		$\odot$	Smear Location				
01	✓	1/2	1.11 $\times 10^{-12}$		$\triangle$	Air Sample Location				
01	✓	1/2	4.5 $\times 10^{-13}$		$\square$	Large Area Smear				
02					*	Contact Reading				
02					Other	1/2				

BD-6000-010 (06/96)

HHC - Radiological Control RADIOLOGICAL SURVEY REPORT			F.C.	Page <u>2</u> of <u>3</u>	Survey No. <u>R000060</u>							
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS						
			h (non-pen) mrad/h	h (pen) mR/h	h (non-pen) mrem/h	Total (/100 cm ²)		Removable (/100cm ²)				
						g (dpm)	g (dpm)	g (dpm)	g (dpm)	mrad/h		
11	INSIDE TUBE 1049	1/2							1000	140	1/2	
12	TUBE 1049 CAP	1/2							1000	140	1/2	
13	BALL PINS	1/2							<D	<D	1/2	
14	STEP OFF PAD	1/2							<D	<D	1/2	
15	PERSONNEL	1/2							<D	<D	1/2	
16	MISC TOOLS & SMARS (MIS)	1/2							3000	350	1/2	
		1/2										
Comments												
TOTAL <D = <u>2</u> inches/sec REMOVAL <D = <u>1</u> inches/sec												
<u>0.25</u> inches from item <u>0.25</u> inches from item												
<u>100</u> % of item surveyed <u>100</u> % of item surveyed												
No audible counts above background No audible counts above background												
* ALL SMARS WERE LARGE AREA WIPES.												
* NON-REM READING IS WIPED OPEN READING WITH CP AND NO CORRECTION FACTOR.												
* ITEM 16 MISC TOOLS & SMARS WERE GATHERED FOR LAB ANALYSIS.												
* PORT WAS COVERED WITH PLYWOOD AND PLASTIC.												
* CAMS FOR JGS WERE SET AT 10% OF DCL FOR PM AND SP.												
* SAMPLE ON ITEM 18 WAS HIGHEST READING. SMAR COULD NOT BE REPEATED AT LEVEL FOUND.												
Samples Counted with portable instruments for <u>0.25</u> minutes.												
Instr./Probe Model	RO-3B (CP)	E-140/Pancake (GM/Probe)	Surveyor-X w/ZnS probe (PAM)									
Serial No.	1683-0049	CMERF-0049	ACHM2-0047									
Efficiency	100%	10%	14%									
Correction Factor	1	10	7									
ADDITIONAL REPORTS												
RFI No.	Sample Counter Log(s)			R-0111467-219								
RSR No.	1/2	Contamination - Skin			1			Procedure: 1/2				
Contamination - Clothing		1/2	Other(s)			1/2						
HPT:	Payroll No.: 45337			First-Line HP Manager:								
Name (Print):	C. WILDMAN			Date: 1/23/97				Payroll No.: 64783				
Signature:	C. WILDMAN			Name (Print): D. R. ECKSTROM				Signature: D. R. Eckstrom				

BD-5000-010R (06/96)

[illegible]

Westinghouse Hanford Company - Radiological Control RADIOLOGICAL SURVEY REPORT				Survey No. R 000063																																																																																																																																													
Date 01/22/1997		Time 0900-1500		RWP No.(s) R-193																																																																																																																																													
Area/Bldg./Room/Location (Code) 300 AREA / 309 Bldg. / O'LEUR CONTAINMENT / UNDER ROTATING SHIELD				Page 1 of 5																																																																																																																																													
Job Description PERFORM CHARACTERIZATION OF PROCESS TUBES (1550, 1354, 1144)				F.C. 2																																																																																																																																													
Purpose of Survey (check appropriate box(es)): Contamination Incident: ☐ Skin, ☐ Clothing, ☐ Spill Alarm Response: ☐ CAM, ☐ ARM(RAM), ☐ PSD ☐ HRA/VHRA Work ☐ Job Coverage ☒ Other ☐ Exposure Incident ☐ Material Release ☐ RM Transfer/Shipments ☐ Required, Task No.																																																																																																																																																	
Map/Sketch 																																																																																																																																																	
<table border="1" style="width: 100%; border-collapse: collapse;"> <thead> <tr> <th rowspan="2">No.</th> <th rowspan="2">Description</th> <th rowspan="2">Dist.</th> <th colspan="3">DOSE RATES</th> <th colspan="5">CONTAMINATION LEVELS</th> </tr> <tr> <th>Σ (non-pen) mrad/h</th> <th>γ (pen) mR/h</th> <th>α mrem/h</th> <th colspan="2">Total (1/100 cm²)</th> <th colspan="3">Removable (1/100cm²)</th> </tr> <tr> <th></th> <th></th> <th></th> <th>μdpm</th> <th>dpm</th> <th>μdpm</th> <th>dpm</th> <th>μdpm</th> <th>dpm</th> <th>mrad/h</th> <th></th> </tr> </thead> <tbody> <tr> <td>1</td> <td>TUBE 1550 CAP</td> <td>1/2</td> <td></td> <td></td> <td></td> <td></td> <td></td> <td>4000</td> <td><3</td> <td>1/2</td> </tr> <tr> <td>2</td> <td>TUBE 1550</td> <td>C</td> <td>1/2</td> <td>3.5</td> <td>1/2</td> <td></td> <td></td> <td>5000</td> <td><3</td> <td>1/2</td> </tr> <tr> <td>3</td> <td>TUBE 1550 (MAX) DOSE/LEAK</td> <td>F</td> <td>1/2</td> <td>1878</td> <td>1/2</td> <td></td> <td></td> <td></td> <td></td> <td></td> </tr> <tr> <td>4</td> <td>TUBE 1354 CAP</td> <td>1/2</td> <td></td> <td></td> <td></td> <td></td> <td></td> <td>4500</td> <td><3</td> <td>1/2</td> </tr> <tr> <td>5</td> <td>TUBE 1354</td> <td>C</td> <td>1/2</td> <td>3.7</td> <td>1/2</td> <td></td> <td></td> <td>10000</td> <td><3</td> <td>1/2</td> </tr> <tr> <td>6</td> <td>TUBE 1354 (MAX) DOSE/LEAK</td> <td>F</td> <td>1/2</td> <td>1640</td> <td>1/2</td> <td></td> <td></td> <td></td> <td></td> <td></td> </tr> <tr> <td>7</td> <td>TUBE 1144 CAP</td> <td>1/2</td> <td></td> <td></td> <td></td> <td></td> <td></td> <td>4000</td> <td><3</td> <td>1/2</td> </tr> <tr> <td>8</td> <td>TUBE 1144</td> <td>C</td> <td>1/2</td> <td>3.5</td> <td>1/2</td> <td></td> <td></td> <td>4000</td> <td><3</td> <td>1/2</td> </tr> <tr> <td>9</td> <td>TUBE 1144 (MAX) DOSE/LEAK</td> <td>F</td> <td>1/2</td> <td>1550</td> <td>1/2</td> <td></td> <td></td> <td></td> <td></td> <td></td> </tr> <tr> <td>10</td> <td>BULL PEN FLOOR</td> <td>1/2</td> <td></td> <td></td> <td></td> <td></td> <td></td> <td><3</td> <td><3</td> <td>1/2</td> </tr> </tbody> </table>						No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS					Σ (non-pen) mrad/h	γ (pen) mR/h	α mrem/h	Total (1/100 cm ²)		Removable (1/100cm ²)						μdpm	dpm	μdpm	dpm	μdpm	dpm	mrad/h		1	TUBE 1550 CAP	1/2						4000	<3	1/2	2	TUBE 1550	C	1/2	3.5	1/2			5000	<3	1/2	3	TUBE 1550 (MAX) DOSE/LEAK	F	1/2	1878	1/2						4	TUBE 1354 CAP	1/2						4500	<3	1/2	5	TUBE 1354	C	1/2	3.7	1/2			10000	<3	1/2	6	TUBE 1354 (MAX) DOSE/LEAK	F	1/2	1640	1/2						7	TUBE 1144 CAP	1/2						4000	<3	1/2	8	TUBE 1144	C	1/2	3.5	1/2			4000	<3	1/2	9	TUBE 1144 (MAX) DOSE/LEAK	F	1/2	1550	1/2						10	BULL PEN FLOOR	1/2						<3	<3	1/2
No.	Description	Dist.	DOSE RATES						CONTAMINATION LEVELS																																																																																																																																								
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Continued on page 2																																																																																																																																																	
Air Sample Results (μCi/ml)				Legend																																																																																																																																													
BZ	GA	Initial	Decay	① - Smear Location	△ - Air Sample Location																																																																																																																																												
#1	✓	1/2	1.0 x 10 ⁻¹⁵	② - Large Area Smear	* - Contact Reading																																																																																																																																												
#1	✓	1/2	1.5 x 10 ⁻¹²	Other 1/2																																																																																																																																													
#2	1/2																																																																																																																																																
#2																																																																																																																																																	

BD-8000-010 (06/95)

WHC - Radiological Control RADIOLOGICAL SURVEY REPORT				F.C.	Page <u>2</u> of <u>5</u>		Survey No. <u>R000063</u>			
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS				
			H (non-pen) mrad/h	Y (pen) mR/h	N mrem/h	Total (/100 cm ²)		Removable (/100cm ²)		
						g (dpm)	g (dpm)	g (dpm)	g (dpm)	mrad/h
11	STEP OFF PAD	1/4						<D	<D	<1/4
12	BETTESOR	1/4				<D	<D	<D	<D	<1/4
13	PERSONNEL	1/4				<D	<D	<1/4		
		1/4								
<div style="position: absolute; top: 0; left: 0; width: 100%; height: 100%; background: linear-gradient(to top right, transparent 49%, black 49%, black 51%, transparent 51%); background-size: 100% 100%; pointer-events: none;"></div>										
Comments										
TOTAL <D = <u>2</u> inches/sec REMOVABLE <D = <u>1</u> inches/sec <u>0.25</u> inches from item (SWAPS) <u>0.25</u> inches from item <u>100</u> % of item surveyed <u>100</u> % of item surveyed No audible counts above background No audible counts above background * SEE ATTACHED SHEETS FOR MORE INFORMATION * ITEMS 2,5,8 WERE TAKEN WITH CP * ITEMS 3,4,9 WERE TAKEN WITH 20-7 * TECH SAMPLES TAKEN WILL BE SENT TO LAB FOR ANALYSIS.										
Samples Counted with portable instruments for <u>0.35</u> minutes.										
Instr./Probe Model	RO-3B (CP)	E-140/Paneco (GM/Probel)	Surveyor-X w/ZnS probe (PAM)	20-7	1/4					
Serial No.	1CEB3-0145	2CEB5-0084 BCEB5-0087	ACUB3-0087	1CEB2-0036						
Efficiency	100%	10%	14%	100%						
Correction Factor	1	10	7	1						
ADDITIONAL REPORTS										
RPB No.	<u>1</u>	Sample Counter (Log/s)	<u>R-0133197-219</u>							
RSR No.	<u>1/4</u>	Contamination - Skin	<u>1/4</u> Procedure: <u>1/4</u>							
Contamination - Clothing		<u>1/4</u>	Other(s)							
HPT:	Payroll No.: <u>49837</u>									
Name (Print): <u>C. Williams</u>		First-Line HP Manager: Date: <u>1/28/97</u> Payroll No.: <u>6A783</u>								
Signature: <u>[Signature]</u>		Name (Print): <u>D.R.</u> Signature: <u>[Signature]</u>								

DATA SHEET 2, TUBE SURVEY

	LEVEL	TUBE	TUBE	TUBE
		1550	1354	1144
	CP READINGS			
	0"	3.5 mr/hr	3.7 mr/hr	3.5 mr/hr
	CONTAMINATION INSIDE PROCESS TUBE			
	TECH SMEAR	5,000 DPM $\beta\gamma$ <20 DPM α	10,000 DPM $\beta\gamma$ <20 DPM α	4,000 DPM $\beta\gamma$ <20 DPM α
	R0-7 READINGS (using low range probe)			
	0"	20 mr/hr	20 mr/hr	0 mr/hr
TOP SHIELD	-24"	80	60	60
	-44"	166	130	84
	-64"	1000	1000	1000
REACTOR VESSEL	-72"	1878	1640	1484
	-84"	1677	1525	1430
	-96"	1733	1551	1480
	-108"	1779	1620	1550
	-120"	1762	1601	1530
	-132"	1655	1533	1450
	-144"	1525	1435	1316
	-156"	1383	1315	1297
	-168"	1315	1225	1150
	-180"	1625	1345	1077
BOTTOM SHIELD	-192"	196	110	100

SURVEY # R000063

PAGE 3 OF 5

C. WILDMAN 1/22/97

DATA SHEET 1

- 1550 (1) 5" OD SHROUD TUBE
 (84) 4 1/4" OD SHROUD TUBES
 (18) 2 9/16" OD SHIM CONTROL SHROUD TUBES
 (13) 1 1/2" OD MONITOR TUBES
 (13) 1" NPT CALANDRIA ACCESS OPENINGS

INFORMATION ONLY

SURVEY # R000063 PAGE 4 OF 5

02/16 1/26/97

Westinghouse Hanford Company - Radiological Control RADIOLOGICAL SURVEY REPORT						Survey No. R 000064				
Date 01/23/1997		Time 1230-1300		RWP No.(s) R-183		Page 1 of 3				
Area/Bldg./Room/Location (Code) 300 AREA 305 B26						F.C. R				
Job Description CHARACTERIZATION OF PROCESS TUBE # 1049 AND SHIELD REPLACEMENT, DOWNGRADE BULL PEN AFTER SHIELD REPLACEMENT				Purpose of Survey (check appropriate box(es)): Contamination Incident: ☐ Skin, ☐ Clothing, ☐ Spill Alarm Response: ☐ CAM, ☐ ARM(RAM), ☐ PSD ☐ HRA/VHRA Work ☒ Job Coverage ☐ Other ☐ Exposure Incident ☐ Material Release ☐ RM Transfer/Shipment ☐ Required, Task No.						
Map/Sketch A sketch of the area with a diagonal line from the top left to the bottom right, labeled 'A' at the top left.										
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS				
			Σ (non-pen) mrad/h	γ (pen) mR/h	α mrem/h	Total I/100 cm ²		Removable I/100cm ²		
			g (dpm)	g (dpm)	g (dpm)	g (dpm)	g (dpm)	g (dpm)	g (dpm)	mrad/h
1	PAGE EXPERIMENT REMOVED	C	1/4	1	5	1/4		<D	<D	1/4
2	PROCESS TUBE 1049 CLOSED	1/4						2000	<D	1/4
3	ROTATING SHIELD REPLACED	1/4						<D	<D	1/4
4	TOOLS & EQUIPMENT	1/4				<D	<D	<D	<D	1/4
5	BULL PEN	1/4						<D	<D	1/4
6	USED LAUNDRY BAG	C	1/2	<0.5	1/4			<D	<D	1/4
7	WASTE BAG	C	1/2	<0.5	1/4			<D	<D	1/4
8	STOP OFF PAD	1/4				<D	<D	<D	<D	1/4
		1/4								
Continued on page 2										
Air Sample Results (uCi/m)					Legend					
BZ	GA	Initial	Decay		(P) - Smear Location	(A) - Air Sample Location				
#1	1/4				(S) - Large Area Smear	* - Contact Reading				
#2					Other 1/4					
#3										

BD 6000-010 (05/95)

[illegible]

BD-6000-010R (06/96)

300 AREA 309 BLDG.**UNDER ROTATING SHIELD TECH SMEARS
UPPER SECTION OF REACTOR CORE**

FOR ANALYSIS AT 325 LAB

Perform GROSS ALPHA, GROSS BETA, GEA on smears

PRTR-C-01	CORE FLOOR	500 DPM $\beta\gamma$	<20 DPM α
PRTR-C-02	FLOOR BETWEEN PROCESS TUBES	1,000 DPM $\beta\gamma$	<20 DPM α
PRTR-C-03	SHIELD PLUG PORT	500 DPM $\beta\gamma$	<20 DPM α
PRTR-C-04	PROCESS TUBES TOP	500 DPM $\beta\gamma$	<20 DPM α
PRTR-C-05	PROCESS TUBE CAPS	500 DPM $\beta\gamma$	<20 DPM α
PRTR-C-06	TUBE SIDES	1,000 DPM $\beta\gamma$	<20 DPM α
PRTR-C-07	TUBE OUTLET PIPES	500 DPM $\beta\gamma$	<20 DPM α
PRTR-C-08	CAVITY CEILING A	500 DPM $\beta\gamma$	<20 DPM α
PRTR-C-09	CAVITY CEILING B	500 DPM $\beta\gamma$	<20 DPM α

PRTR-PT-01	PROCESS TUBE # 1550	5,000 DPM $\beta\gamma$	<20 DPM α
PRTR-PT-02	PROCESS TUBE # 1354	10,000 DPM $\beta\gamma$	<20 DPM α
PRTR-PT-03	PROCESS TUBE # 1144	4,000 DPM $\beta\gamma$	<20 DPM α

Several of the smears are on whatman 5 filter papers. While others were taken on stickly backed smear papers and stuck to a whatman 5 filter paper.

Readings were taken with a EGM and a PAM for reference only.

Contact RCT Chuck Wildman at 376-5115 for any questions about the smears.

R.000064 PAGE 3 of 3

1/23/97

Westinghouse Hanford Company - Radiological Control RADIOLOGICAL SURVEY REPORT						Survey No. R000089					
Date <u>02/19/99</u>		Time <u>0800-1500</u>		RWP No.(s) <u>R-184 REV 001</u>		Page <u>1</u> of <u>4</u>					
Area/Bldg./Room/Location (Code) <u>30CA26A 309 Bldg. "C" Cell - 21' Level</u>						F.C. <u>R</u>					
Job Description <u>REMOVE ACCESS PLUG DISPOSE</u> <u>OF ION CHAMBER AND CHARACTERIZE</u> <u>PLUG PORT</u>				Purpose of Survey (check appropriate box(es)):							
				Contamination Incident: ☐ Skin, ☐ Clothing, ☐ Spill Alarm Response: ☐ CAM, ☐ ARM(IRAM), ☐ PSD ☐ HRA/VHRA Work ☒ Job Coverage ☐ Other ☐ Exposure Incident ☐ Material Release ☐ RM Transfer/Shipments ☐ Required, Task No. _____							
Map/Sketch 											
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS					
			$\dot{H}$ (unpen) mrad/h	$\dot{H}$ (pen) mR/h	$\dot{H}$ mrem/h	Total (/100 cm ²)		Removable (/100cm ²)			
						$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	mrad/h
1	FIRST STEP OF PLUG								15000	<3	—
2	SECOND STEP OF PLUG								100000	<3	—
3	RE-SMEAR STEP								20000	<3	—
4	THIRD STEP OF PLUG								10000	<3	—
5	RE-SMEAR STEP								40000	<3	—
6	PLUG PORT WITH CP	C	—	3.5	—						
7	PLUG PORT 1ST 6" OF PORT								250000	<3	2
8	TECH SMEAR 2ND 6" OF PORT								60000	<3	0.5
9	PLUG PORT IN 6" WITH CP	F	—	8	—						
10	WORK AREA	F	—	<0.5	—				<3	<3	—
Continued on page 2											
Air Sample Results (μ Ci/ml)						Legend					
	BZ	GA	Initial	Decay		☒ - Smear Location - Air Sample Location ☒ - Large Area Smear • - Contact Reading Other <u>—/—</u>					
01	—	—									
01	—	—									
02	—	—									
02	—	—									

BD-5000-010 (06/95)

WHC - Radiological Control RADIOLOGICAL SURVEY REPORT				F.C. R	Page 2 of 4	Survey No. R000089						
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS						
			α (non-pen) mrad/h	γ (pen) mR/h	n mrem/h	Total (/100 cm ²)		Removable (/100cm ²)				
						α (dpm)	β (dpm)	α (dpm)	β (dpm)	mrad/h		
11	Fission Chamber	C	—	<0.1	—	—	—	—	—	5000	<β	—
12	Chamber Cut & Bagged	C	—	<0.1	—	—	—	—	—	<β	<β	—
13	TELE MAX READING in Port	F	—	9000	—	—	—	—	—	—	—	—
14	Work Area	—	—	—	—	—	—	—	—	<β	<β	—
15	STOP OFF PAD	—	—	—	—	—	—	—	—	<β	<β	—
16	PERSONNEL	—	—	—	—	<β	<β	—	—	—	—	—
Comments * CD = 1"/sec 0.25" FROM ITEM 100% OF SURVEY (NO AUDIBLE COUNTS ABOVE BACKGROUND) * CD TOTAL = 2"/sec 0.25" FROM ITEM 100% OF ITEM SURVEY (NO AUDIBLE COUNTS ABOVE BACKGROUND) * SEE PAGE 3 FOR DRAWING OF PLUG & THIMBLE * SEE PAGE 4 FOR TELETECTOR READINGS INSIDE THIMBLE (PORT) * FISSION CHAMBER BODY WERE CUT INTO 2 SECTIONS / RODS AND CHAMBER WAS WRAPPED IN PLASTIC, TAPPED AND PLACED IN 55 GAL DRUM FOR DISPOSAL. * PLUG WAS WRAPPED IN PLASTIC AND SET TO THE SIDE. * PLUG PORT WAS COVERED WITH PLASTIC AT END OF DAY. * CANS IN ROOM WERE SET TO ALARM AT 10% OF DAC * ALL SMEARS WERE LARGE AREA WIPES UNLESS NOTED. Samples Counted with portable instruments for 0.25 minutes.												
Instr./Probe Model	RO-3B (CP)	E-140/Pencok (GMI/Probe)	Surveyor-X w/ZnS probe (PAM)	TELE	HA							
Serial No.	1CER3-0496	CMR88-0328	ACQUZ-0103	XYXT3-0008	HA							
Efficiency	100%	10%	14%	100%								
Correction Factor	1	10	7	1								
ADDITIONAL REPORTS												
RPR No.	Sample Counter Log(s)											
RSR No.	HA	... Contamination - Skin				HA	Procedure: HA					
... Contamination - Clothing HA Other(s)												
HPT:	Payroll No.: 9563											
Name (Print): C. WILBANE					First-Line HP Manager: Date: 2-24-97 Payroll No.: 95641							
Signature: C. WILBANE					Name (Print): G. A. DAY							
					Signature: G. A. DAY							

BD-6000-010R (06/96)

309 BLDG. -21' Level "C" Cell Plug

* 60,000 DPM B/G
* <20 DPM Alpha
TECH SMEAR

* 250,000 DPM B/G
* <20 DPM Alpha
LARGE AREA WIPE

Secondary Smears
* 2-40,000 DPM B/G
* <20 DPM Alpha

* 100,000 DPM B/G
* <20 DPM Alpha

* 100,000 DPM B/G
* <20 DPM Alpha

* 15,000 DPM B/G
* <20 DPM Alpha

STEP PLUG

100-10-0000 000000

- Fixed Contamination
- ★ Removable Contamination

HPT CAO, Date 2/18/97

SURVEY # R000089 PAGE 3 OF 4

-21' Level "C" Cell Step Plug Thimble

Estimated distances in thimble

Dose readings in mr/hr

4	@	opening
7	@	1st Step
14	@	2nd Step
25	@	Tube opening
48	@	1' further
180	@	2' further
1000	@	3' further
9000	@	4' further
8500	@	5' further
9000	@	Tube end

Estimate Distance used Teletector and estimated distance in Thimble

Dose reading with CP 3.5 mr/hr @ Opening

8 mr/hr @ 6" in opening

SURVEY # R000089

PAGE 4 of 4

C. WILDMAN

02/18/1997

Westinghouse Hanford Company - Radiological Control RADIOLOGICAL SURVEY REPORT						Survey No. R000091				
Date <u>02/19/1997</u>		Time <u>1000-1400</u>		RWP No.(s) <u>R-184 REV 001</u>		Page <u>1</u> of <u>4</u>				
Area/Bldg./Room/Location (Code) <u>300 AREA / 309 ALCL / 21' LEVEL "C" CPU</u>						F.C. <u>R</u>				
Job Description <u>PERFORM CHARACTERIZATION SURVEY</u> <u>OF PLUG PORT AND CONTACT GAMMA ENERGY</u> <u>ANALYSIS</u>				Purpose of Survey (check appropriate box(es)): Contamination Incident: ☐ Skin, ☐ Clothing, ☐ Spill Alarm Responses: ☐ CAM, ☐ ARM(RAM), ☐ PSD ☐ HRA/VHRA Work ☒ Job Coverage ☐ Other ☐ Exposure Incident ☐ Material Release ☐ RM Transfer/Shipmt ☐ Required, Task No.						
<div style="text-align: right; margin-bottom: 5px;">Map/Sketch</div>										
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS				
			α (non-pen) mrad/h	γ (pen) mR/h	n mrem/h	Total (1/100 cm ²)		Removable (1/100cm ²)		
						μ (dpm)	e (dpm)	μ (dpm)	e (dpm)	mrad/h
1	MAX DOSE IN THIMBLE	F	—	1100	—	—	—	—	—	—
2	WORK AREA WHOLE BODY	F	—	D.S	—	—	—	—	—	—
3	DETECTORS	—	—	—	—	<5	<5	<5	<5	—
4	WORK AREA	—	—	—	—	—	—	<5	<5	—
5	STEP OFF PAD	—	—	—	—	—	—	<5	<5	—
6	PERSONNEL	—	—	—	—	<5	<5	—	—	—
7	SEE ATTACHED PAGES	—	—	—	—	—	—	—	—	—
			Continued on page 2							
Air Sample Results (μ Ci/ml)					Legend					
	BZ	GA	Initial	Decay	☒ - Smear Location	☒ - Air Sample Location				
#1	—	—	—	—	☒ - Large Area Smear	* - Contact Reading				
#2	—	—	—	—	Other <u>—</u>					
#2	—	—	—	—						

BD-8000-010 (06/96)

WHC - Radiological Control RADIOLOGICAL SURVEY REPORT						F.C. _____ Page <u>2</u> of <u>4</u>	Survey No. <u>R000091</u>				
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS					
			h (non-pen) mrad/h	h (pen) mR/h	h microm/h	Total (1/100 cm ²)		Removable (1/100 cm ²)			
						g (dpm)	g (dpm)	g (dpm)	g (dpm)	g (dpm)	mrad/h
1											
<div style="display: flex; justify-content: space-between;"> <div> <p>Comments</p> <p>TOTAL $\leq D = 2$ inches/sec</p> <p><u>0.25</u> inches from item</p> <p><u>100</u> % of item surveyed</p> <p>No audible counts above background</p> </div> <div> <p>REMOVABLE $\leq D = 1$ inches/sec</p> <p><u>0.25</u> inches from item</p> <p><u>100</u> % of item surveyed</p> <p>No audible counts above background</p> </div> </div> <p>* SEE ATTACHED PAGES FOR MORE INFORMATION.</p> <p>* DETECTORS WERE BAGGED IN PLASTIC PRIOR TO ENTRY IN THIMBLE AND PLASTIC WAS REMOVED ON EXIT.</p> <p>* FOOT WAS COVERED WITH PLASTIC AT END OF DAY</p> <p>* CAMS SET AT 10% OF DAC FOR ALARM POINT.</p> <p>* ALL SURFACES WERE LARGE AREA WIPES.</p> <p>Samples Counted with portable instruments for <u>0.25</u> minutes.</p>											
Instr./Probe Model	RO-3B (CP)	E-140/Pancake (GM/Probe)	Surveyor-X w/ZnS probe (PAM)	TELETYPE	RO-7/2840C 200K TMR	RO-7/2840C 200K TMR	RO-7/2840C 200K TMR	RO-7/2840C 200K TMR	RO-7/2840C 200K TMR	RO-7/2840C 200K TMR	RO-7/2840C 200K TMR
Serial No.	1CEB-0484	CEB-0318	ACH-0111	XXXX-0008	1CEB-0036	1CEB-0036	1CEB-0036	1CEB-0036	1CEB-0036	1CEB-0036	1CEB-0036
Efficiency	100%	10%	14%	100%	100%	100%	100%	100%	100%	100%	100%
Correction Factor	1	10	7	1	1	1	1	1	1	1	1

ADDITIONAL REPORTS

RPR No. _____ Sample Counter Log(s) _____

RSR No. 1A Contamination - Skin _____ Procedure: 1A

Contamination - Clothing 1A Other(s) _____

HPT: _____

Payroll No.: 95837

Name (Print): C. WILKINSON

Signature: [Signature]

First-Line HP Manager: _____

Date: 2-21-97 Payroll No.: 95641

Name (Print): G.A. DALL

Signature: [Signature]

309 BLDG. -21' Level "C" Cell Plug

SEP-19-01 10:00 AM

- * Fixed Contamination
- * Removable Contamination

HPT 0-10, Date 2/19/97

SURVEY # 2000041 PAGE 4 OF 4

HN-SD-NEL-ER-008 REV 0

Westinghouse Hanford Company - Radiological Control RADIOLOGICAL SURVEY REPORT					Survey No. R000092					
Date 02/20/1997		Time 0800-1000		RWP No.(s) R-184 REV001		Page 1 of 2				
Area/Bldg./Room/Location (Code) 300 AREA / 309 BLDG / -21' LEVEL "C" CELL						F.C. R				
Job Description REPLACE PLUG INTO WALL AND REMOVE RALCA				Purpose of Survey (check appropriate box(es)):						
				Contamination Incident: ☐ Skin, ☐ Clothing, ☐ Spill						
				Alarm Response: ☐ CAM, ☐ ARM(RAM), ☐ PSD						
				☐ HRA/VHRA Work ☒ Job Coverage ☐ Other						
				☐ Exposure Incident ☐ Material Release						
				☐ RM Transfer/Shipments ☐ Required, Task No.						
Map/Sketch										
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS				
			# (non-pen) mrad/h	γ (pen) mR/h	α mrem/h	Total (1/100 cm ²)		Removable (1/100cm ²)		
						g (dpm)	μ (dpm)	g (dpm)	μ (dpm)	mrad/h
1	PLUG REPLACED @ Port					4000		<D	<D	
2	WALLS							<D	<D	
3	FLOOR							<D	<D	
4	EQUIPMENT & TOOLS					<D	<D	<D	<D	
5	PERSONNEL					<D	<D			
6	LAUNDRY BGL (water ppr)	C		<0.5				<D	<D	
7	S&P WASTE BGL	C		<0.5				<D	<D	
8	STEP OFF RAD							<D	<D	
			Continued on page 2							
Air Sample Results (μCi/ml)					Legend					
	BZ	GA	Initial	Decay	①	②				
# 1	N/A				① - Smear Location	② - Large Area Smear				
# 2					⚠ - Air Sample Location	* - Contact Reading				
# 3					Other <u>N/A</u>					

Westinghouse Hanford Company - Radiological Control RADIOLOGICAL SURVEY REPORT					Survey No. R000095						
Date 02/24/1997		Time 1100-1500		RWP No.(s) R-183		Page 1 of 4					
Area/Bldg./Room/Location (Code) 300 AREA 309 Bldg CONTAINMENT REACTOR CAVITY						F.C. R					
Job Description PERFORM CHARACTERIZATION OF REACTOR CAVITY THROUGH PROCESS TUBES				Purpose of Survey (check appropriate box(es)): Contamination Incident: ☐ Skin, ☐ Clothing, ☐ Spill Alarm Response: ☐ CAM, ☐ ARM(IRAM), ☐ PSD ☐ HRA/VHRA Work ☐ Job Coverage ☒ Other ☐ Exposure Incident ☐ Material Release ☐ RM Transfer/Shipiment ☐ Required, Task No.							
<div style="display: flex; justify-content: space-between;"> Map/Sketch 1/A </div>											
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS					
			β (non-pen) mrad/h	γ (pen) mR/h	α mrem/h	Total (100 cm ²)		Removable (100cm ²)			
			g (dpm)	μ (dpm)	g (dpm)	μ (dpm)	mrad/h				
1	ROTATING SHIELD PLUG							<5	<5		
2	PLUG PORT	F		1				<5	<5		
3	PROCESS TUBE CAPS	C		2				1000	<5		
4	TUBE 1550 CAP REMOVED	C		3.5				3000	<5		
5	TUBE (MAX WITH RO-7)	C		44700							
6	TUBE 1324 CAP REMOVED	C		3.5				3000	<5		
7	TUBE (MAX WITH RO-7)	C		42200							
8	TUBE 1144 CAP REMOVED	C		3.5				5000	<5		
9	TUBE (MAX WITH RO-7)	C		26000							
10	DETECTOR CABLE							<5	<5	<5	<5
Continued on page 2											
Air Sample Results (uCi/ml)						Legend					
	BZ	GA	Initial		Decay	①	Smear Location				
						②	Air Sample Location				
						③	Large Area Smear				
						④	Contact Reading				
1	1/A					Other 1/A					
2											
3											
4											
5											
6											
7											
8											
9											
10											

BD-6000-010 (06/96)

WHC - Radiological Control RADIOLOGICAL SURVEY REPORT				F.C.	Page <u>2</u> of <u>4</u>	Survey No. R000095				
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS				
			β (non-pen) mrad/h	γ (pen) mR/h	α (mem/h)	Total (1/100 cm ²)		Removable (1/100cm ²)		
						g (dpm)	g (dpm)	g (dpm)	g (dpm)	mrad/h
11	GULL PEN	F	—	40.5	—	—	—	—	<D	<D
12	PROCESS TUBE 1049 CAP	C	—	3.5	—	—	—	4000	<D	—
13	GULL PEN	—	—	—	—	—	—	—	<D	<D
14	STOP OFF DAD	—	—	—	—	—	—	—	<D	<D
h/k										
Comments										
TOTAL <D = <u>2</u> inches / sec REMOVABLE <D = <u>1</u> inches / sec <u>0.25</u> inches from item <u>0.15</u> inches from item <u>100</u> % of item surveyed <u>100</u> % of item surveyed No audible counts above background No audible counts above background * ALL SURFACES WERE WASHED WITH WIPES / TUBES 1550, 1354, 1114 WERE CLOSED UP AFTER READINGS WERE TAKEN WITH RO-7 (SEE ATTACHED DATA SHEET 2) * TUBE / GA EXPIRIMENT WAS PLACED IN TUBE 1049 AND WILL BE REMOVED 2/23/97 * PLUG PORT WAS COVERED WITH PLYWOOD WHILE TEST IN PLUG PORT OVERNIGHT. * CAMS ALARM SET POINT WAS 10% OF DAC FOR PU-239 97-90. -/k										
Samples Counted with portable instruments for <u>0.25</u> minutes.										
Instr./Probe Model	RO-3B (CP)	E-140/Pancake (GM/Probe)	Surveyor-X with probe (PAM)	40-7 low range probe no range mode						
Serial No.	1CEB3-046L	KMEB3-0313 BTMC-0103	ACMM2-0202	1CEB3-0036 VTR-0036 VTR-0036						
Efficiency	100%	10%	14%	100%						
Correction Factor	1	10	7	1						
ADDITIONAL REPORTS										
RPR No.	Sample Counter Log(s)									
RSR No.	Contamination - Skin									
Contamination - Clothing										
Other(s)										
HPT:	First-Line HP Manager:									
Payroll No.: 95837	Date: 2-26-97 Payroll No.: 95641									
Name (Print): C. L. DAVIS	Name (Print): G. A. DAVIS									
Signature: C. L. DAVIS	Signature: G. A. DAVIS									

BD-5000-010R (06/96)

DATA SHEET 2, TUBE SURVEY

	LEVEL	TUBE		TUBE		TUBE	
		1550		1354		1144	
	CP READINGS						
	0"	3.5 mr/hr		3.5 mr/hr		3.5 mr/hr	
	R0-7 READINGS						
	LEVEL	LR	MR	LR	MR	LR	MR
	0"	3	0	3	0	4	0
	TOP SHIELD	-24"	10	0	12	0	11
	-44"	92	200	89	200	53	100
	-64"	--	38500	--	34900	--	19300
MAX DOSE	-66"	--	46700	--	42200	--	26000
REACTOR VESSEL	-72"	--	19700	--	16100	--	12900
	-84"	--	14600	--	13000	--	11400
	-96"	--	15400	--	13500	--	12300
	-108"	--	16800	--	14700	--	13300
	-120"	--	17100	--	14900	--	13600
	-132"	--	15900	--	14000	--	12800
	-144"	--	13800	--	12500	--	11000
	-156"	--	11600	--	11000	--	9300
	-168"	--	10100	--	9300	--	8100
	-180"	--	13500	--	11200	--	7600
BOTTOM SHIELD	-192"	1975	700	1245	600	1000	200
	-204"	30	0	28	0	15	0

INSTRUMENT RO-7 ICEB2-0036; DTEB6-0030 LR Probe; DTEB7-0047 MR Probe

LR=LOW RANGE PROBE 0-2R / MR=MID RANGE PROBE 0-200R

(--)=Beyond Probe Range, LR Probe 1 mr/hr increments / MR Probe 100 mr/hr increments

SURVEY R000095 PAGE 3 OF 4

02/24/1997 C. WILDMAN

Westinghouse Hanford Company - Radiological Control RADIOLOGICAL SURVEY REPORT						Survey No. R 000096				
Date <u>02/25/1997</u>		Time <u>1230-1500</u>		RWP No.(s) <u>R-183</u>		Page <u>1</u> of <u>4</u>				
Area/Bldg./Room/Location (Code) <u>300 AREA / 309 RIDGE / CONTAINMENT</u>						F.C. <u>R</u>				
Job Description <u>PERFORM CHARACTERIZATION SURVEY</u> <u>OF PROCESS TUBE AND REMOVE EXPERIMENT</u> <u>FROM REACTIVE COIL.</u> <u>W/</u>				Purpose of Survey (check appropriate boxes): Contamination Incident: ☐ Skin, ☐ Clothing, ☐ Spill Alarm Response: ☐ CAM, ☐ ARM(RAM), ☐ PSD ☐ HRA/VHRA Work ☐ Job Coverage ☒ Other ☐ Exposure Incident ☐ Material Release ☐ RM Transfer/Shipments ☐ Required, Task No. _____						
Map/Sketch 										
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS				
			$\dot{H}$ (non-pen) mrad/h	$\dot{H}$ (pen) mR/h	$\dot{H}$ mrem/h	Total (/100 cm ²)		Removable (/100cm ²)		
			$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	$\dot{H}$ (dpm)	$\dot{H}$ (dpm)
1	BULL PEN							<D	<D	—
2	EXPERIMENT REMOVED							<D	<D	—
3	CLOSE TUBE 1049	F		1				1000	<D	—
4	REMOVE CAP AND TUBE 1550							2000	<D	—
5	MAX DOSE RATE IN TUBE (RO-7)	F		44300						—
6	RO-7 DETECTOR AND CABLE					<D	<D	<D	<D	—
7	CLOSE TUBE 1550	F		1				1000	<D	—
8	ROTATING SHIELD REPLACED	F		<0.5				<D	<D	—
9	BULL PEN							<D	<D	—
10	STEP OFF PAD							<D	<D	—

Continued on page 2

Air Sample Results (μ Ci/ml)				Legend	
BZ	GA	Initial	Decay	(S) - Smear Location	(A) - Air Sample Location
#1	#1			(F) - Large Area Smear	(C) - Contact Reading
#1				Other <u>W/</u>	
#2					
#2					

BD-6000-010 (06/96)

WHC - Radiological Control RADIOLOGICAL SURVEY REPORT				F.C.	Page <u>2</u> of <u>4</u>	Survey No. R 000096			
No.	Description	Dist.	DOSE RATES			CONTAMINATION LEVELS			
			■ (non-pen) mrad/h	γ (pen) mR/h	γ mrem/h	Total (/100 cm ²)		Removable (/100cm ²)	
						■ (dpm)	■ (dpm)	■ (dpm)	mrad/h
11	USED LAUNDRY BAG	C	—	<0.5	—	—	—	<D	<D
12	SOP WASTE BAG	C	—	<0.5	—	—	—	<D	<D
13	MISC EQUIPMENT	—	—	—	—	<D	<D	<D	<D
<div style="position: relative; width: 100%; height: 100%;"> <div style="position: absolute; top: 0; left: 0; width: 100%; height: 100%; border: 1px solid black; transform: rotate(45deg); transform-origin: top left;"></div> </div>									
Comments									
TOTAL <D = 2 inches / sec REMOVABLE <D = 1 inches / sec <u>0.25</u> inches from item <u>0.25</u> inches from item <u>50</u> % of item surveyed <u>100</u> % of item surveyed No audible counts above background No audible counts above background									
* ALL SHEETS WERE LARGE AREA WIPES. BULK PEN WAS DOWNGRADED AFTER COMPLETION OF JOB WASTE AND USED LAUNDRY WERE PLACED IN RM's. * CAM'S WERE ALARM SET POINT AT 10% OF DAC FOR Pu239, Sr90 * SEE ATTACHED DATA SHEET FOR INFORMATION TAKEN FROM THIS ISSO. * PING TO FUDGE JOINT WAS TAPED OVER AND COVERED WITH RUC TO PROTECT. * BULK PEN WAS MOVED TO NEXT JOB LOCATION.									
Samples Counted with portable instruments for <u>0.25</u> minutes.									
Instr./Probe Model	RO-3B (CP)	E-140/Pancake (GM/Probe)	Surveyor-X w/ZnS probe (PAM)	RO-7 Increase					
Serial No.	10EB3-0486	CANES-0378 BTANC-0103	ACHW-0102	10E02-0036	10E07-0047				
Efficiency	100%	10%	14%	100%					
Correction Factor	1	10	7	1					
ADDITIONAL REPORTS									
RPR No. <u>1</u> Sample Counter Log(s) <u>1</u> ASR No. <u>1</u> ... Contamination - Skin <u>1</u> Procedure: <u>1</u> ... Contamination - Clothing <u>1</u> Other(s) <u>1</u>									
HPT: Payroll No.: <u>95837</u>					First-Line HP Manager: Date: <u>2-26-97</u> Payroll No.: <u>95641</u>				
Name (Print): <u>C. Williams</u>					Name (Print): <u>CA Davis</u>				
Signature: <u>[Signature]</u>					Signature: <u>[Signature]</u>				

DATA SHEET 3, TUBE SURVEY

LEVEL	TUBE
	1550
R0-7 READINGS (using mid range probe)	
-44"	300
-48"	600
-52"	1800
-56"	8800
-58"	15600
-60"	23900
-62"	32800
-64"	40300
-66"	44300
-68"	28100
-70"	22000
-72"	19500

ICEB2-0036 RO-7
 DTEB7-0047 Mid Range Probe
 Mid Range Probe 0-200R
 100 mr/hr increments for probe

SURVEY R000096
 PAGE 3 OF 4
 02/25/1997
 C. WILDMAN

Battelle Pacific Northwest Laboratory
Analytical Chemistry Laboratory
Radioanalytical Group - 325 Bldg.

97-1696
3/4/97

Client : D. Cornwell
Wp#: K76109

Cognizant Scientist:

V. K. Fiddell

Date :

3/4/97

Concur :

T. Treadwell

Date :

3/4/97

ALO ID Client ID	Measured Activities (uCi/sa)				
	Total Alpha Error %	Total Beta Error %	Co-60 Error %	Cs-137 Error %	Am-241 Error %
97-1696 *	5.64E-8	7.13E-5	4.14E-5	3.18E-5	<7.E-6
PRTR-C-01	12%	5%	5%	7%	
97-1697 *	5.38E-5	1.34E-4	7.05E-5	5.64E-5	5.16E-5
PRTR-C-02	5%	4%	5%	6%	10%
97-1698	1.39E-5	3.10E-6	<3.E-6	<3.E-6	<7.E-6
PRTR-C-03	19%	14%			
97-1699	1.38E-6	3.45E-5	1.52E-5	9.96E-6	<7.E-6
PRTR-C-04	60%	11%	8%	13%	
97-1700	2.49E-7	5.30E-6	6.38E-6	<3.E-6	<2.E-5
PRTR-C-05	47%	10%	14%		
97-1701	<2.E-7	8.48E-6	4.44E-6	<3.E-6	<7.E-6
PRTR-C-06		8%	20%		
97-1702	4.30E-7	7.60E-6	7.61E-6	<3.E-6	<2.E-5
PRTR-C-07	35%	8%	11%		
97-1703	1.09E-5	1.37E-5	2.75E-6	2.67E-6	<6.E-6
PRTR-C-08	7%	6%	27%	39%	
97-1704	3.01E-7	5.21E-7	<3.E-6	<3.E-6	<7.E-6
PRTR-C-09	45%	52%			
97-1705	6.03E-6	6.63E-4	1.05E-3	3.99E-6	<3.E-5
PRTR-PT-01	28%	4%	5%	48%	
97-1706	4.98E-5	5.74E-3	7.04E-3	<4.E-5	<5.E-5
PRTR-PT-02	10%	3%	5%		
97-1707	7.82E-6	7.13E-4	1.04E-3	<2.E-5	<4.E-5
PRTR-PT-03	25%	4%	5%		
Standard	99%	103%			
Blank	<9.E-8	5.79E-7 20%			

Note: * PRTR-C-01 and PRTR-C-02 required acid digestion due to high sample activity.
The standard and blank results are from the QC samples run with their preparation.

-32' LEVEL LOWER ACCESS TECH SMEARS

FOR ANALYSIS AT 325 LAB

Perform **GROSS ALPHA, GROSS BETA, GEA** on smears

PRTR-SP-0A	FLOOR UNDER DUMP TANK	4000 DPM $\beta\gamma$	140 DPM α
PRTR-SP-0B	FLOOR UNDER DUMP TANK	3000 DPM $\beta\gamma$	140 DPM α
PRTR-SP-0C	PIPE TO DUMP TANK	2000 DPM $\beta\gamma$	<20 DPM α
PRTR-SP-0D	TANK SUMP BOTTOM SIDE	40000 DPM $\beta\gamma$	420 DPM α

PRTR-TK-01	DUMP TANK INSIDES BOTTOM	15000 DPM $\beta\gamma$	700 DPM α
------------	--------------------------	-------------------------	------------------

PRTR-LA-01	WALL OF LOWER ACCESS	1000 DPM $\beta\gamma$	<20 DPM α
PRTR-LA-02	FLOOR/ENTRANCE TO LA	2000 DPM $\beta\gamma$	<20 DPM α
PRTR-LA-03	TOP DRAIN LINE PIPE	15000 DPM $\beta\gamma$	<20 DPM α
PRTR-LA-04	BOTTOM GAS DUMP VALVE	2000 DPM $\beta\gamma$	<20 DPM α
PRTR-LA-05	CEILING OF LOWER ACCESS	1000 DPM $\beta\gamma$	<20 DPM α

Several of the smears are on whatman 5 filter papers. While others were taken on stickly backed smear papers and stuck to a whatman 5 filter paper.

Readings were taken with a EGM and a PAM for reference only.

Contact RCT Chuck Wildman at 376-5115 for any questions about the smears.

300 AREA 309 BLDG.**UNDER ROTATING SHIELD TECH SMEARS
UPPER SECTION OF REACTOR CORE**

FOR ANALYSIS AT 325 LAB

Perform **GROSS ALPHA, GROSS BETA, GEA** on smears

PRTR-C-01	CORE FLOOR	500 DPM $\beta\gamma$	<20 DPM α
PRTR-C-02	FLOOR BETWEEN PROCESS TUBES	1,000 DPM $\beta\gamma$	<20 DPM α
PRTR-C-03	SHIELD PLUG PORT	500 DPM $\beta\gamma$	<20 DPM α
PRTR-C-04	PROCESS TUBES TOP	500 DPM $\beta\gamma$	<20 DPM α
PRTR-C-05	PROCESS TUBE CAPS	500 DPM $\beta\gamma$	<20 DPM α
PRTR-C-06	TUBE SIDES	1,000 DPM $\beta\gamma$	<20 DPM α
PRTR-C-07	TUBE OUTLET PIPES	500 DPM $\beta\gamma$	<20 DPM α
PRTR-C-08	CAVITY CEILING A	500 DPM $\beta\gamma$	<20 DPM α
PRTR-C-09	CAVITY CEILING B	500 DPM $\beta\gamma$	<20 DPM α

PRTR-PT-01	PROCESS TUBE # 1550	5,000 DPM $\beta\gamma$	<20 DPM α
PRTR-PT-02	PROCESS TUBE # 1354	10,000 DPM $\beta\gamma$	<20 DPM α
PRTR-PT-03	PROCESS TUBE # 1144	4,000 DPM $\beta\gamma$	<20 DPM α

Several of the smears are on whatman 5 filter papers. While others were taken on sticky backed smear papers and stuck to a whatman 5 filter paper.

Readings were taken with a EGM and a PAM for reference only.

Contact RCT Chuck Wildman at 376-5115 for any questions about the smears.

Battelle Pacific Northwest Laboratory
Analytical Chemistry Laboratory
Radioanalytical Group - 325 Bldg.

97-1597
2/10/97

Client: B. Cornwell
Vop#: K76109

Cognizant Scientist: J.R. Greenwood

Date: 2/10/97

Concur: J. Truong

Date: 2/10/97

Measured Activities (uCi/sa)

ALO ID Client ID	Total Alpha Error %	Total Beta Error %	Co-60 Error %	Cs-137 Error %	Am-241 Error %
97-1597	2.41E-4	7.67E-4	1.26E-5	5.37E-4	6.75E-5
PRTR-SP-OA	13%	8%	7%	5%	7%
97-1598	1.66E-4	3.18E-4	5.45E-6	1.39E-4	3.56E-5
PRTR-SP-OR	12%	15%	10%	5%	7%
97-1599	3.57E-5	1.86E-4	4.25E-6	9.75E-5	1.45E-5
PRTR-SP-OC	27%	20%	11%	2%	8%
97-1600	7.07E-4	7.58E-3	4.56E-5	7.42E-3	2.87E-4
PRTR-SP-OD	8%	9%	3%	2%	3%
97-1601	2.59E-4	1.84E-3	2.73E-3	2.04E-5	6.57E-5
PRTR-TK-01	9%	22%	2%	15%	8%
97-1602	<9 E-6	<5 E-5	<2 E-6	1.51E-6	<6 E-6
PRTR-LA-01				49%	
97-1603	<1 E-5	1.15E-4	5.70E-6	6.28E-5	1.99E-6
PRTR-LA-02		18%	8%	3%	40%
97-1604	1.33E-5	2.14E-3	3.67E-6	8.34E-4	<9 E-6
PRTR-LA-03	56%	18%	22%	5%	
97-1605	<9 E-6	1.26E-4	4.23E-6	9.83E-5	7.88E-5
PRTR-LA-04		29%	10%	2%	12%
97-1606	<9 E-6	<5 E-5	<1 E-6	4.77E-6	<3 E-6
PRTR-LA-05				12%	
Standard	85%	90%			
Blank	<1 E-6	<6 E-6			

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APPENDIX G
SURFACE AREA CALCULATIONS AND ASSUMPTIONS

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Assumptions and Methods Used to Estimate Surface Areas

Complex shapes were broken down into as many simple shapes as needed to make a reasonable approximation. For example, the inside of the reactor vessel was calculated as a cylinder with a complete end (rather than subtracting out the areas occupied by the process tube shrouds) filled with cylinders (the shrouds). If a particular component didn't look like it would add substantially to the surface area it was ignored. An example of this, again from the reactor vessel, was to ignore the shim rod shrouds and neutron monitor shrouds.

Primary Coolant Surface Area

Inlet Ring Header - a torus, 13' diameter and 10" thick.

$$A = \pi^2 Dd, \text{ where } D = \text{diameter and } d = \text{thickness}$$

$$\text{if } D = 13' \text{ and } d = 0.83'$$

$$\text{Area} = 106 \text{ ft}^2 \text{ or } 9.89\text{E}4 \text{ cm}^2$$

Outlet Ring Header - torus, 13.5' diameter and 10" thick.

$$A = \pi^2 Dd, \text{ where } D = \text{diameter and } d = \text{thickness}$$

$$\text{if } D = 13.5' \text{ and } d = 0.83'$$

$$\text{Area} = 111 \text{ ft}^2 \text{ or } 1.03\text{E}5 \text{ cm}^2$$

Header Inlet and Outlet Pipes, one for each header (2), assumed to be identical.

$$A = 2(\pi DL), \text{ where } D = \text{diameter and } L = \text{length}$$

$$\text{if } D = 14'' \text{ and } L = 7',$$

$$\text{Area} = 51.5 \text{ ft}^2 \text{ or } 4.78\text{E}4 \text{ cm}^2$$

Process Tubes, cylinders, assumed to be same diameter over entire length.

$$A = n(\pi DL), \text{ where } n = \text{number of tubes, } D = \text{pipe diameter, and } L = \text{length}$$

$$\text{if } n = 85, D = 3.25'' (0.27'), \text{ and } L = 17'5'' (17.4')$$

$$\text{Area} = 1,254 \text{ ft}^2 \text{ or } 1.17\text{E}6 \text{ cm}^2$$

Outlet Nozzles, Cylinders, ends were ignored.

$$A = n(\pi DL), \text{ where } n = \text{number of tubes, } D = \text{pipe diameter, and } L = \text{length}$$

$$\text{if } n = 85, D = 3.25'' (0.27'), \text{ and } L = 2'$$

$$\text{Area} = 144 \text{ ft}^2 \text{ or } 1.34\text{E}5 \text{ cm}^2$$

Jumpers, assumed to be straight cylinders, of same length, average 10' long, inlet and outlet.

$$A = n(\pi DL), \text{ where } n = \text{number of tubes, } D = \text{pipe diameter, and } L = \text{length}$$

$$\text{if } n = 85, D = 1.5'' (0.13'), \text{ and } L = 10'$$

$$\text{Area} = 694 \text{ ft}^2 \text{ or } 6.45\text{E}5 \text{ cm}^2$$

Total Primary Surface Area=2.36E3 ft² or 2.19E6 cm²

Moderator Surface Area

Tank Inside - cylinder capped on both ends

$$A = 2(\pi(1/2D)^2) + \pi DL, \text{ where } D = \text{tank diameter, } L = \text{tank height} \\ \text{if } L = 6'3" (6.25'), D = 15' \\ \text{Area} = 648 \text{ ft}^2 \text{ or } 6.02\text{E}5 \text{ cm}^2$$

Tank Sump - cylinder, open on top

$$A = \pi(1/2D)^2 + \pi DL, \text{ where } D = \text{sump diameter, } L = \text{sump depth} \\ \text{if } D = 2' \text{ and } L = 2' \\ \text{Area} = 15.7 \text{ ft}^2 \text{ or } 1.46\text{E}4 \text{ cm}^2$$

Top Gas Lines - cylinders, open on both ends

$$A = n(\pi DL), \text{ where } D = \text{pipe diameter, } L = \text{pipe length, and } n = \text{number of pipes} \\ \text{if } D = 10" (.833'), L = 24', \text{ and } n = 4 \\ \text{Area} = 251 \text{ ft}^2 \text{ or } 2.33\text{E}5 \text{ cm}^2$$

Bottom Gas Lines - cylinders, open on both ends

$$A = n(\pi DL), \text{ where } D = \text{pipe diameter, } L = \text{pipe length, and } n = \text{number of pipes} \\ \text{if } D = 10" (.833'), L = 15', \text{ and } n = 4 \\ \text{Area} = 157 \text{ ft}^2 \text{ or } 1.46\text{E}5 \text{ cm}^2$$

Lower Wier Baffle Plate - (this is the upward turned portion of the inlet plenum wier that separates the inlet plenum from the dump chamber, as well as its supporting plate) assume cylinder, both sides, negligible material thickness, closed at one end

$$A = 2(\pi DL) + 2\pi(1/2D)^2, \text{ where } D = \text{cylinder diameter and } L = \text{cylinder height} \\ \text{if } D = 8'8" (8.67') \text{ and } L = 1'5" (1.42') \\ \text{Area} = 195 \text{ ft}^2 \text{ or } 1.82\text{E}5 \text{ cm}^2$$

Dump Chamber, Calandria Bottom Plate, and Upper Weir Baffle - this is the area directly under the reflector tank, encompassing the entire calandria bottom plate. Assume upper weir baffle is flat.

$$A = \pi(1/2D)^2 + \pi DL + (\pi(1/2D)^2 - \pi(1/2d)^2), \text{ where } D = \text{calandria diameter,} \\ L = \text{dump tank height, and } d = \text{reactor vessel diameter} \\ \text{if } D = 10.83', d = 8.67', \text{ and } L = 2' \\ \text{Area} = 193 \text{ ft}^2 \text{ or } 1.80\text{E}5 \text{ cm}^2$$

Vessel Wall - the boundary wall between the reactor vessel and the reflector tank, assume cylinder with no holes

$$A=CL, \text{ where } C=\text{circumference and } L=\text{height}$$

$$\text{if } C=22' \text{ and } L=7'$$

$$\text{Area} = 154 \text{ ft}^2 \text{ or } 1.43\text{E}5 \text{ cm}^2$$

Center Process Tube Shroud - uncapped cylinder

$$A=\pi DL, \text{ where } D=\text{pipe diameter and } L=\text{pipe length}$$

$$\text{if } D=6" (.5') \text{ and } L=9'9" (9.75')$$

$$\text{Area} = 15.3 \text{ ft}^2 \text{ or } 1.42\text{E}4 \text{ cm}^2$$

Other Process Tube Shrouds - uncapped cylinders

$$A=n(\pi DL), \text{ where } n=\text{number of tubes, } D=\text{pipe diameter, } L=\text{pipe length}$$

$$\text{if } n=84, D=4.25" (.354'), \text{ and } L=9'9" (9.75')$$

$$\text{Area} = 911 \text{ ft}^2 \text{ or } 8.46\text{E}5 \text{ cm}^2$$

Shim Tubes - uncapped cylinders

$$A=n(\pi DL), \text{ where } n=\text{number of tubes, } D=\text{pipe diameter, } L=\text{pipe length}$$

$$\text{if } n=19, D=2 \frac{9}{16}" (.214'), \text{ and } L=9'9" (9.75')$$

$$\text{Area} = 125 \text{ ft}^2 \text{ or } 1.16\text{E}5 \text{ cm}^2$$

Monitor Tubes - uncapped cylinders

$$A=n(\pi DL), \text{ where } n=\text{number of tubes, } D=\text{pipe diameter, } L=\text{pipe length}$$

$$\text{if } n=13, D=1.5" (.125'), \text{ and } L=9'9" (9.75')$$

$$\text{Area} = 49.8 \text{ ft}^2 \text{ or } 4.62\text{E}4 \text{ cm}^2$$

Top Plate - ring, like a washer

$$A= \pi(1/2D)^2 - \pi(1/2d)^2, \text{ where } D= \text{greater diameter and } d= \text{lesser diameter}$$

$$\text{if } D=11' \text{ and } d= 7'$$

$$\text{Area} = 56.5 \text{ ft}^2 \text{ or } 5.25\text{E}4 \text{ cm}^2$$

Top Chamber - cylinder capped on one end

$$A= \pi(1/2D)^2 + \pi DL, \text{ where } D= \text{calandria diameter, and } L= \text{chamber height}$$

$$\text{if } D=11' \text{ and } L= 8" (.667')$$

$$\text{Area} = 118 \text{ ft}^2 \text{ or } 1.10\text{E}5 \text{ cm}^2$$

Top Drain - uncapped cylinder

$$A= \pi DL, \text{ where } D= \text{pipe diameter, and } L= \text{pipe length}$$

if $D=10''$ (.833') and $L=23.75'$
Area = 62.2 ft^2 or 5.77E4 cm^2

Bottom Drain - uncapped cylinder

$A = \pi DL$, where D = pipe diameter, and L = pipe length
if $D=10''$ (.833') and $L=14'$
Area = 36.6 ft^2 or 3.40E4 cm^2

Total Moderator Surface Area = 2.99E3 ft^2 or 2.78E6 cm^2

Dry Gas Space

Inside surface of Biological Shield - a series of different-sized cylinders, ledges ignored.

A = bottom cylinder + calandria cylinder + header cylinder + top cylinder, or
 $A = \pi D_b L_b + \pi D_c L_c + \pi D_h L_h + \pi D_t L_t$
if $D_b = 11'1''$ (11.08'), $L_b = 1'$
 $D_c = 12'8''$ (12.67'), $L_c = 11.2'$
 $D_h = 13'8''$ (13.67'), $L_h = 1'1''$ (1.08')
 $D_t = 15'$, $L_t = 1.92'$
Area = 617 ft^2 or 5.74E5 cm^2

Thermal Shield - outside surface of thick cylinder

$A = 2[\pi(1/2D_o)^2 - \pi(1/2D_i)^2] + \pi D_o L + \pi D_i L$, where D_o = outside diameter, D_i = inside diameter, and L = height of thermal shield
if $D_o = 12'6''$ (12.5'), $D_i = 11'6''$ (11.5'), and $L = 12'$
Area = 942 ft^2 or 8.76E5 cm^2

Bottom Shield - assume flat top, treat as cylinder capped on one end

$A = \pi(1/2D)^2 + \pi DL$, where D = shield diameter, L = shield height
if $D=11'4''$ (11.3'), $L=3'4''$ (3.33')
Area = 219 ft^2 or 2.03E5 cm^2

Calandria - assume a cylinder capped on both ends

$A = 2[\pi(1/2D)^2] + \pi DL$, where D = calandria diameter, L = calandria height
if $D=11'$, $L=9'11''$ (9.92')
Area = 533 ft^2 or 4.95E5 cm^2

Top Shield - assume cylinder capped on one end

$A = \pi(1/2D)^2 + \pi DL$, where D = shield diameter, L = shield height
if $D=14'10''$ (14.83'), $L=(1.92')$
Area = 262 ft^2 or 2.44E5 cm^2

Shroud Tubes - from Moderator calculation

$$\text{Area} = 926 \text{ ft}^2 \text{ or } 8.61\text{E}5 \text{ cm}^2$$

Process Tubes - from Primary calculation

$$\text{Area} = 1,254 \text{ ft}^2 \text{ or } 1.17\text{E}6 \text{ cm}^2$$

$$\text{Total Dry Gas Surface Area} = 4.75\text{E}3 \text{ ft}^2 \text{ or } 4.42\text{E}6 \text{ cm}^2$$

Reflector Tank Surface

- assume simple annular tank

$A = 2[\pi(1/2D_o)^2 - \pi(1/2D_i)^2] + \pi D_o L + \pi D_i L$, where D_o = outside diameter, D_i = inside diameter, and L = height of thermal shield

if $D_o = 11'$, $D_i = 7'$, and $L = 7.08'$

$$\text{Area} = 513 \text{ ft}^2 \text{ or } 4.77\text{E}5 \text{ cm}^2$$

Top Face

Top of Top Shield - assume flat circle with capped cylinder in the middle

$A = \pi(1/2D_o)^2 + \pi D_i L$, where D_o = diameter of shield, D_i = diameter of inside raised portion, and L = height of raised portion

if $D_o = 14'$, $D_i = 11.5'$, and $L = 1.08'$

$$\text{Area} = 193 \text{ ft}^2 \text{ or } 1.79\text{E}5 \text{ cm}^2$$

Nozzles - from earlier calculation (Primary Coolant surface)

$$\text{Area} = 144 \text{ ft}^2 \text{ or } 1.34\text{E}5 \text{ cm}^2$$

Jumpers - from earlier calculation (Primary Coolant surface)

$$\text{Area} = 347 \text{ ft}^2 \text{ or } 3.23\text{E}5 \text{ cm}^2$$

Outlet Ring Header - from earlier calculation (Primary Coolant surface)

$$\text{Area} = 111 \text{ ft}^2 \text{ or } 1.03\text{E}5 \text{ cm}^2$$

Outside wall - assume cylinder (note: bottom of rotating shields NOT included since they were non-smearable)

$A = \pi D L$, where D = diameter of top face chamber, L = height of chamber

if $D = 15.5'$ and $L = 3.58'$

$$\text{Area} = 174 \text{ ft}^2 \text{ or } 1.62\text{E}5 \text{ cm}^2$$

$$\text{Total Area for Top Face} = 9.69\text{E}2 \text{ ft}^2 \text{ or } 9.11\text{E}5 \text{ cm}^2$$

Under Moderator Tank

- assume cylinder capped on both ends

$A = 2[\pi(1/2D)^2] + \pi D L$, where D = chamber diameter and L = chamber height

if $D = 18'6"$ (18.5') and $L = 3'$

$$\text{Area} = 712 \text{ ft}^2 \text{ or } 6.61\text{E}5 \text{ cm}^2$$

Bottom Face - the lower reactor space above the thermal barrier

Bottom Face space - assume cylinder capped on both ends

$$A = 2[\pi(1/2D)^2] + \pi DL, \text{ where } D = \text{chamber diameter and } L = \text{chamber height}$$

if $D = 18'6"$ (18.5') and $L = 4'$
 Area = 770 ft² or 7.15E5 cm²

Inlet Ring Header - from Primary Coolant surface calculations

$$\text{Area} = 106 \text{ or } 9.89\text{E4 cm}^2 \text{ ft}^2$$

Ring Header Inlet - from Primary Coolant surface calculations

$$\text{Area} = 25.7 \text{ ft}^2 \text{ or } 2.39\text{E4 cm}^2$$

Jumpers - from Primary Coolant surface calculations

$$\text{Area} = 347 \text{ ft}^2 \text{ or } 3.22\text{E5 cm}^2$$

Total Area for Bottom Face 1.25E3 ft² or 1.16E6 cm²

Lower Reactor Space - ignore thermal barrier lift, tools, and thermal barrier ring support

Walls and Ceiling - assume cylinder capped on one end, include temp barriers

$$A = \pi(1/2D)^2 + \pi DL, \text{ where } D = \text{room diameter and } L = \text{room height}$$

if $D = 18'6"$ (18.5') and $L = 7'$
 Area = 676 ft² or 6.28E5 cm²

Floor - assume circle

$$A = \pi(1/2D)^2, \text{ where } D = \text{diameter of the room}$$

if $D = 18.5'$
 Area = 269 ft² or 2.50E5 cm²

Pipes - assume cylinders

$$A = n(\pi DL), \text{ where } n = \text{number of pipes, } D = \text{diameter of pipes, and } L = \text{Length of pipes}$$

if $n = 10$, $D = 10"$ (0.83'), and $L = 7'$
 Area = 183 ft² or 1.70E4 cm²

Total Area for Lower Reactor Space = 1.13E3 ft² or 1.05E6 cm²

APPENDIX H
OLD SURVEYS

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ATTACHMENT - RADIATION SURVEY REPORT #700149

Tube measured at one foot increments as withdrawn from core.

Reading #1, tube measured through $\frac{1}{4}$ " steel, TP probe unshielded.

Reading #2, tube measured through $\frac{1}{4}$ " steel, TP probe shielded top and bottom

Reading #3, tube measured through $\frac{1}{4}$ " steel and $\frac{1}{4}$ " lead, TP probe shielded top and bottom with $\frac{1}{4}$ " lead.

<u>LEVEL</u>	<u>#1 READING</u>	<u>#2 READING</u>	<u>#3 READING</u>
0 foot	Bkgd = 50 mR/hr		
1 foot	75 "		
2 foot	160 "		200 mR/hr
3 foot	700 "		600 "
4 foot	1700 "		1400 "
5 foot	4100 "		2700 "
6 foot	7500 "		4100 "
7 foot	10,000 "		4900 "
8 foot	12,000 "		5500 "
9 foot	11,000 "	10,000 mR/hr	4500 "
foot	9500 "	8000 "	3700 "
foot	7000 "	6500 "	2800 "
12 foot	5000 "		

All measurements made with TP probe.
12 inches from ϕ of tube. Shielding 6 inches
from ϕ of tube

Imeehs

BLDG. 309	LOCATION Top of Reactor	TIME 0900 TO 1130	DATE 12/4/70	SURVEY LOG NO 700149
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[illegible]

INSTRUMENT TYPE					RESPIRATORY PROTECTION REQUIRED		
☐ EGM	☐ TGM	☐ POPPY	☒ CP	☐ BFG	☐ SUPPLIED AIR	☐ FILTER	☐ NONE
☐ HGM	☐ MW-GM	☐ JUND	☒ TPC	☐ SCINTRAN			

PERSONNEL DOSE RATES

PHASE OF WORK	NO.	WHOLE BODY PEN. mm/hz	WHOLE BODY SKIN mm/hz.	EXTREMITIES mm/hz
Operate Crane and Make Measurements	6	100		

COMMENTS
See Attachments

Effort was made to shield TP probe so measurements were of radiation through shielding only.

Shielding
TP Probe
TP Probe Shielding

MAX. UNCORRECTED MEASUREMENT 12,000 mR/hr		MAX. CORRECTED DOSE RATE 10X mR/hr		SIGNATURE FL Meeks <i>FL Meeks</i>	
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ATTACHMENT - RADIATION SURVEY REPORT #700143

Measure Process Tube from PWR Core

Dose rate under reactor a very even and uniform 300 mr/hr. Measurement taken for exposure to hand @ 2' - 3' below pigtail tubes. No measurement above background detected @ angle valves and dump valves. Dose rate atop tube caps 50 mr/hr for personnel kneeling or squatting atop caps. Dose rate to 150 mr/hr beside tubes @ flanges.

Measurements of process tube made with CP tube positioned 12" from E at tube at Reactor Hall floor level. First measurement made when tube flange was raised on floor level. CP measurements @ 8'9" from E of tube.

	CP @ 1'	TP @ 1'	CP @ 8'9"
Flange @ Floor Level	100		
1'	240	300	5
2'	600	700	15
3'	1500	1700	50
4'	3200	3600	150
5'	These meas.	7700	250
6'	@ same position	12000	350
7'	of TP tube	16000	400
8'		10000	600
9'		16000	600
10'		14000	700
11'		9000	700
12'		6000	700

100	Director's Office 1000 and 2000	0530 TO 1230	DATE 12/2/70	FILE NO. OR NO. 700148
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NO.	ITEM OF LOCATION MEASUREMENT	METER DIFF.		SOURCE SIZE	DIST.	BETA	GAMMA	NEUTRON	DIRECT BEAMS
		WO	WC			INSTR.	CH/IN	CH/IN	CM OR IN/CM OR IN
1.	Average D/R on top caps	50	50	7 5"	2000		50		
2.	Av. D/R on lower access	300	300	7 5"	2000		150		
3.	Control Spill wells (max)	100/150	2050	7 6"	2"		2000		
4.	D/R @ angle valves	100/150	150	7 5"	3		150		
5.	D/R on walkway around lower access	100	100	7 5"	1		100		
6.	Beams of lower access shield								1500 d/r 1000 d/r
7.	Wipe (max)			15% of tube	10"				1000 d/r
8.	Internal smear of tube								1500 cpm d/r
9.	Internal smear of tube	50	5						100 cpm d/r

INSTRUMENT TYPE					RESPIRATORY PROTECTION REQUIRED	
☐ EGM	☐ TGM	☐ POPPY	☒ CP	☐ BFO	☐ SUPPLIED AIR	☒ FILTER
☐ HCM	☐ MW-HCM	☐ WING	☐ ALTPC	☐ SCINTRAN		☐ NONE

PHASE OF WORK	NO.	WHOLE BODY PEN. AREA/HR	WHOLE BODY SKIN AREA/HR	EXTREMITIES AREA/HR
Remove top cap	1	50		
Remove Bellows	2	300		

ACTION TAKEN

 $\gamma_1 = \gamma_2 = \dots = \gamma_n = \gamma$

APPENDIX I
RADION-CHEMISTRY REPORT ON GAMMA
AND NUETRON DATA

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Pacific Northwest Laboratories
Battelle Boulevard
P.O. Box 999
Richland, Washington 99352
Telephone (509) 376-0405

May 8, 1997

Bruce C. Cornwell
BWHC
Mail Stop L5-70
Richland, WA 99352

Dear Mr. Cornwell:

PNNL DOWN-HOLE ASSAYER RESULTS

Enclosed is a set of data for work performed on various dates between **January 22, 1997**, and **February 24, 1997**, using the PNNL Down-Hole Assayer System at **2 locations** inside the decommissioned Plutonium Recycle Test Reactor (PRTT) core inside the **309 Building, 300 Area**. The assay was intended to document the main radioisotopes contributing to the gamma-ray dose levels inside the core and to confirm the levels of residual transuranic (TRU) isotopes that may have settled on surfaces of the core following the fuel failure that terminated the useful role of the reactor.

The PNNL Down-Hole Assayer System is a non-destructive assay (NDA) system that employs high-resolution (HPGe) gamma-ray spectrometry and passive neutron counting to characterize a local radiological environment with confined space or where physical access is restricted.

Activity Measured via Gamma-Ray Spectrometry

The results of the gamma-ray analysis for the reactor core was complicated by the small size of the access ports, which, in turn, limited the amount of shielding that could be used to reduce instrumental dead time. An additional complication to the assay was the pervasive use of steel for both structural support and for radiation shielding; the net result was a very high level of the activation product ⁶⁰Co in all measurements, especially at points where a high neutron fluence would have existed, such as the headspace at the top of the moderator volume. The other radioisotopes measured were the fission products ¹³⁷Cs and ¹⁵²Eu, which appear to be present in the process tubes that made up the core proper.

B. C. Cornwell
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Alpha-Particle Activity Measured via Passive Neutron Counting

The neutron count for the PNNL Down-Hole Assay System is taken separately from acquisition of gamma-ray spectra. The neutron count is performed with a single $^{10}\text{BF}_3$ neutron counting tube with 5-cm diameter and 1.83 m length. The neutron count is typically taken with both a bare tube and with the tube wrapped in polyethylene that provides moderation for higher energy neutrons in the source term, so that some information regarding the energy spectrum of the neutron source is obtained. The neutron counting tubes used in the PNNL Down-Hole Assay System have been rated to operate in fields up to 5 R/hr without experiencing spurious counts from gamma-ray pulse pile-up.

The neutron count was made only from the thimble that penetrates the reactor wall. Only counts with the unmoderated tube were considered valid since the signal cable disengaged during insertion for the moderated count; the anti-contamination sheath obscured the problem during the count and there was not sufficient time to perform the experiment again. This was not considered to degrade sensitivity significantly because of the moderating effects of water in the concrete surrounding the core.

The neutron tube counts ambient neutrons that are low enough in energy to be captured by the ^{10}B . The counting model assumes two sources of neutrons: one from natural background and one from transuranic (TRU) isotopes. The natural background source of neutrons consists mostly of spallation products caused by cosmic-ray bombardment of surrounding structures, which in this case was depressed because of the shielding effects of the massive concrete overburden. The other source of neutrons is transuranic materials, which has two components associated with it: 1) alpha-particle activity results in neutron emissions via (α, n) reactions, predominantly with oxygen and other light elements of the matrix and 2) spontaneous fission of various transuranic isotopes (typically even-numbered isotopes) releases several neutrons.

When no gamma-rays associated with TRU are directly measured or observed and no process knowledge is provided, then the TRU value determined from the neutron count is based on the specific alpha-particle activity of weapons-grade PuO_2 , a conservative estimate since weapons-grade plutonium yields the lowest alpha-particle activity (hence, the lowest neutron emission rate) of any TRU material that can reasonably be expected in Hanford waste and is an appropriate assumption in this case because of process knowledge.

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Field Calibration Source Check

The gamma-ray spectrometer was checked before use of the instrumentation using a thoria standard. The three peaks at 238.63-, 911.07-, and 2614.47-keV are examined for position and area as part of an operability check and as a control check of the energy and efficiency calibrations. The positions and areas are determined using commercial ADCAM software and compared to previous control values.

For the data taken for this report, no accuracy or precision noncompliance issues exist.

Recommendations and Conclusions

The neutron data show that the PRTR Core contains less than 0.92 g of plutonium within a 95% confidence interval. This value is based on the assumption that any plutonium is broadly dispersed throughout the core surfaces and that the source term behaves as a planar source. Please note that the concrete should behave as a neutron cavity that allows a given neutron multiple chances of detection before being absorbed by materials in the walls; this effect has not been taken into account in this study, but would lower the amount of plutonium that could be present and still escape detection.

Please note that gamma-rays associated with TRU activity would not be detected at these levels using high-resolution gamma-ray spectrometry because of the extremely high Compton-scatter background in the measured spectra that obscure the presence of the relatively low-energy gamma rays associated with TRU.

Also, computer printouts from the code NANAL, used to document these neutron-count data, are provided in Appendix A. One printout provides the background counts taken outside the reactor walls and provides the comparison for a count in similar surroundings with a confirmed lack of TRU. The second printout provides the data obtained with the neutron tube fully inserted inside the thimble.

The predominant fission product measured was ^{137}Cs ; the other fission product measured, ^{152}Eu , appeared to have about 0.001 times the activity of the ^{137}Cs . As shown in the spreadsheet plot in Appendix B, the level of measured activity for ^{137}Cs remained fairly constant in a series of spectra taken at 6-inch intervals beginning at 12 inches inside the concrete wall and ending at 66 inches, at which point instrumental dead time was too high to acquire any further data. Between

B. C. Cornwell

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the 18-inch and 42-inch counting positions, the ratio of activities for $^{60}\text{Co}/^{137}\text{Cs}$ remained constant at roughly 1/170. This ratio is assumed to be the ratio of the two species within the process tube of the moderator portion of the core. The activity of the ^{60}Co increased exponentially beyond this point as the detector face approached the highly activated shielding walls.

Hard copies of the data are enclosed for you to maintain as record copies while floppy diskettes containing the archived spectral data are kept on file for backup purposes at PNNL. Any questions may be directed to me at Mail Stop P8-08, by cc:Mail, by FAX at 376-2329, or by phone at 376-0405.

Sincerely,

Richard J. Arthur
Sr. Research Scientist

Attachments

cc w/attachments:
MD Winterrose, P8-01
File /LB

Appendix A

Computer Printout of NANAL Code Used to
Process and Document Neutron-Count Results

***** 97021901.nan *****

NANAL: Neutron Analysis Code (Version 2.00) Date Analyzed -- (14:18:14 on 05/07/97)

Neutron ANALYSIS file: \DATA\97021901.NAN
 Segment CountTime NeutronCount Discard Sigma from Median
 1 200.00 23. -.94
 2 200.00 32. -.76
 3 200.00 16. -2.27
 4 200.00 19. -1.70
 5 200.00 28. .00
 6 200.00 24. -.76
 7 200.00 29. .19
 8 200.00 28. .00
 9 200.00 38. 1.89
 10 200.00 39. 2.08
 11 200.00 35. 1.32
 & 2200.00 311. 11-Sample Median 28.00

BARREL: 8kg for PRTR Core
 SITE: PRTR: 309 Building; 100 Area

OPERATOR: ndw

Statistics on 11 counts

(Differences in Sigma units)

Segment	Calc Value	Poisson Value	Diff/CalcSD	Diff/PoissonSD
2	5.019835E+01	2.827273E+01	1.44	1.80
3	-3.263722E+01	2.827273E+01	-.44	-.52
4	5.056560E+03	2.426314E+03	1.26	1.04
Mean	28.27	28.27		
SDofMEAN	2.24	1.60		
Std Dev	7.43	5.32	1.97	1.85
SD of SD	1.07	1.14		
Skewness	-.091765	.188069	-.73	-.37
Kurtosis	-.993328	.035370	-2.20	-.61
Median	28.00	28.00	.00	.00
SDof MED	2.81	2.01		

 ** NO NET NEUTRON COUNT **

 ** Measurement error dominates detection efficiency error **
 ** Can profitably increase measurement time by factor of 2.51D+00 **

EFFNEU= 9.01D-03 4.50D-04 5.00%
 Y1= 1.243685 Y2= 7.846913E-01

Only the 11 GOOD neutron counts used in TRU calculations

Measured Neutron Count Rate = 1.41364E-01 c/s +/- 1.12025E-02 [7.92%]
 Estimated Bkgd N Count Rate = 1.41364E-01 c/s +/- 1.12025E-02 [7.92%]
 Net Est Neutron Count Rate = 1.00000E-20 c/s +/- 1.58427E-02 [*****%]
 Measured N Source Strength = 1.11019E-18 n/s +/- 1.75884E-00 [*****%]

PRELIMINARY TRU activity 7.6441E-13 nCi +/- 1.2110E+06 nCi [*****%]
 LESS THAN 1.9921E+06 nCi (1.64-sigma threshold)
 TRU activity Density 7.6427E-16 nCi/gram +/- 1.2108E+03 nCi/gram [*****%]
 LESS THAN 1.9918E+03 nCi/gram based on GROSS WT= 1000. grams
 TRU activity Density 7.6441E-13 nCi/gram +/- 1.2110E+06 nCi/gram [*****%]
 LESS THAN 1.9921E+06 nCi/gram based on NET WT= 1. grams
 Total grams of TRU in barrel 8.6323E-21 gram +/- 1.3676E-02 gram [*****%]
 LESS THAN 2.2497E-02 gram

CALCULATIONS BASED ON: 1.000 kg Net barrel weight 1 N-tubes
 Neutron Efficiency = 9.0075E-03 (4.5037E-04) [cts/s]/[n/s] [5.00%]
 Calibration Factor = 6.8854E+05 (3.4427E+04) nCi/[n/s] [5.00%]
 Specific Activity = 8.8552E+07 (4.0000E+06) nCi/gram [4.52%]
 Meas Bkg Count Rate= 1.4136E-01 (1.1202E-02) cps [7.92%]
 Bkg Gross Barrel Wt= 2.205 # BkgAdjFactor= 1.000000 CFERR= 0.0000E+00
 TRU material assumed to be: Weapons grade Pu oxide

***** 97021903.nan *****

NANAL: Neutron Analysis Code (Version 2.00) Date Analyzed -- (14:35:41 on 05/07/97)

Neutron ANALYSIS file: \DATA\97021903.NAN

Segment	CountTime	NeutronCount	Discard	Sigma from Median
1	200.00	32.	.96	
2	200.00	29.	.38	
3	200.00	25.	-.38	
4	200.00	22.	-.96	
5	200.00	27.	.00	
6	200.00	32.	.96	
&	1200.00	167.	6-Sample Median	27.00

BARREL: Plutonium Recycle Test Reactor Core

SITE: PRTR: 309 Building; 100 Area

OPERATOR: mdw

Statistics on 6 counts

(Differences in Sigma units)

CountMoment	Calc Value	Poisson Value	Diff/CalcSD	Diff/PoissonSD
2	1.313889E+01	2.783333E+01	-3.13	-.91
3	-1.259262E+01	2.783333E+01	-1.45	-.26
4	3.045811E+02	2.351917E+03	-14.43	-.61
Mean	27.83	27.83		
SDofMEAN	1.62	2.15		
Std Dev	3.97	5.28	-2.02	-.85
SD of SD	.65	1.54		
Skewness	-.264410	.189547	-.77	-.45
Kurtosis	-1.235647	.035928	-2.28	-.55
Median	27.00	28.00	-.49	-.37
SDof MED	2.03	2.72		

** NO NET NEUTRON COUNT **

** Measurement error dominates detection efficiency error **

** Can profitably increase measurement time by factor of 2.390+00 **

EFFNEU= 2.16D-04 1.08D-05 5.00%

Y1= 49.946850 Y2= 32.274280

Only the 6 GOOD neutron counts used in TRU calculations

Measured Neutron Count Rate = 1.39167E-01 c/s +/- 1.07690E-02 (7.74%)

Estimated Bkgd N Count Rate = 1.41364E-01 c/s +/- 1.12025E-02 (7.92%)

Net Est Neutron Count Rate = 1.00000E-20 c/s +/- 1.55392E-02 (*****%)

Measured N Source Strength = 4.63800E-17 n/s +/- 7.20711E+01 (*****%)

PRELIMINARY TRU activity 3.1934E-11 nCi +/- 4.9624E+07 nCi (*****%)

LESS THAN 8.1631E+07 nCi (1.64-sigma threshold)

TRU activity Density 3.1929E-14 nCi/gram +/- 4.9615E+04 nCi/gram (*****%)

LESS THAN 8.1616E+04 nCi/gram based on GROSS Wt= 1000. grams

TRU activity Density 3.1934E-11 nCi/gram +/- 4.9624E+07 nCi/gram (*****%)

LESS THAN 8.1631E+07 nCi/gram based on NET Wt= 1. grams

Total grams of TRU in barrel 3.6063E-19 gram +/- 5.6039E-01 gram (*****%)

LESS THAN 9.2184E-01 gram

CALCULATIONS BASED ON: 1.000 kg Net barrel weight 1 N-tubes

Neutron Efficiency = 2.1561E-04 (1.0781E-05) [cts/s]/[n/s] (5.00%)

Calibration Factor = 6.8854E+05 (3.4427E+04) nCi/[n/s] (5.00%)

Specific Activity = 8.8552E+07 (4.0000E+06) nCi/gram (4.52%)

Meas Bkg Count Rate= 1.4136E-01 (1.1203E-02) cps (7.92%)

Bkg Gross Barrel Wt= 3.202 # BkgAdjFactor= 1.0000000 CFERR= 0.0000E+00

TRU material assumed to be: Weapons grade Pu oxide

Appendix B

Spreadsheets Printout Showing Observed Relative
Activities for Cs-137 and Co-60 Inside PRTR Thimble

HNF-SD-NEL-ER-008 REV 0

Date: 3/21/97 2:59 PM

File: 970219PR.XLS

Run No.	Depth (in.)	Co-60 (relative activity)	Cs-137 (relative activity)	Ratio (Cs137/Co60)
97021916	0	1.60E+01	9.54E+02	60
97021906	6	1.85E+01	1.51E+03	82
97021907	12	1.96E+01	7.19E+03	367
97021908	18	3.75E+01	8.15E+03	218
97021909	24	4.15E+01	9.44E+03	228
97021910	30	4.42E+01	5.95E+03	134
97021911	42	3.95E+01	6.61E+03	167
97021912	48	6.17E+01	5.15E+03	83
97021913	54	6.70E+01	5.31E+03	79
97021914	60	1.05E+02	5.24E+03	50
97021915	66	4.69E+02	8.01E+03	17

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APPENDIX J
DOSE RATE PREDICTION CALCULATIONS

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This appendix provides estimates for isotopic concentrations on various PRTR core surfaces. This is done by using information from a sample analyzed during the time the PRTR operated. The samples were reactor fuel with 2.8 yr decay (98% uranium, 2% plutonium). The sample results were back calculated to determine ^{241}Am concentrations. Thirty and sixty yr decay products were calculated from the original isotopes. These calculations are included in Table 1.0. The sixty year projections were made to show relative distribution for a later time, approximately the year 2025.

The thirty year decay results were then "normalized" to ^{241}Am and Total Alpha to allow for isotopic projections based on easily obtained laboratory analysis (i.e. ^{241}Am by Gamma Energy Analysis and Total Alpha by α scintillation count).

Actual sample results were used to project isotopic concentrations for various reactor surfaces. These calculations were done for both americium and total alpha results when available. The two methods of comparison are quite close for surfaces that were not decontaminated. Comparison was not so favorable in areas where decontamination had taken place. These results are reported in Table 2.0 and Table 3.0.

Table 1.0, Isotopic Decay Table

	Specific Activity	2.8 Yr Activity*	Isotopic Content After 30 Years of Decay		Isotopic Content After 60 Years of Decay	
Isotope	Ci/g	Ci/g of fuel	Ci/g of fuel	g isotope/g fuel	Ci/g of fuel	g isotope/g fuel
Am-241	3.4314×10^0	1.7796×10^{-3}	2.63×10^{-3}	7.665×10^{-04}	2.73×10^{-3}	7.956×10^{-4}
Np-237	7.0473×10^8	-	2.26×10^{-8}	3.207×10^{-17}	4.73×10^{-8}	6.712×10^{-17}
Pa-233	2.0800×10^4	-	2.25×10^{-8}	1.082×10^{-12}	4.72×10^{-8}	2.269×10^{-12}
Pa-234m	6.8692×10^8	-	3.27×10^{-7}	4.760×10^{-16}	3.27×10^{-7}	4.760×10^{-16}
Th-231	5.3000×10^5	-	1.43×10^{-8}	2.698×10^{-14}	1.43×10^{-8}	2.698×10^{-14}
Th-234	2.3200×10^4	-	3.27×10^{-7}	1.409×10^{-11}	3.27×10^{-7}	1.409×10^{-11}
U-234	6.2490×10^3	3.1432×10^{-7}	3.23×10^{-7}	5.169×10^{-5}	3.29×10^{-7}	5.265×10^{-5}
U-235	2.1620×10^{-6}	1.4227×10^{-8}	1.43×10^{-8}	6.614×10^{-3}	1.43×10^{-8}	6.614×10^{-3}
U-236	6.4700×10^{-5}	4.5859×10^{-9}	5.02×10^{-9}	7.759×10^{-5}	5.42×10^{-9}	8.377×10^{-5}
U-238	3.3620×10^{-7}	3.2699×10^{-7}	3.27×10^{-7}	9.726×10^{-1}	3.27×10^{-7}	9.726×10^{-1}
Pu-238	1.7120×10^1	1.1280×10^{-4}	8.90×10^{-5}	5.199×10^{-5}	7.12×10^{-5}	4.159×10^{-5}
Pu-239	6.2000×10^{-2}	1.0230×10^{-3}	1.02×10^{-3}	1.645×10^{-2}	1.02×10^{-3}	1.645×10^{-2}
Pu-240	2.2680×10^{-1}	4.8444×10^{-4}	4.83×10^{-4}	2.130×10^{-3}	4.81×10^{-4}	2.121×10^{-3}
Pu-241	1.0300×10^2	3.7106×10^{-2}	8.76×10^{-3}	8.505×10^{-5}	2.25×10^{-3}	2.184×10^{-5}
Pu-242	3.9200×10^{-3}	1.0020×10^{-7}	1.00×10^{-7}	2.551×10^{-5}	1.00×10^{-7}	2.551×10^{-5}

Notes:

- ²⁴¹Am 2.8 yr activity value created by back calculation of ²⁴¹Pu to time zero.
- Original laboratory analysis data is from BNWL-1674, "Plutonium Recycle Burnup Data and Results from the Batch Core Experiment," D.E. Christensen, et.al.

Table 2.0, Isotopic Calculations (Ci/100 cm² sample) by Surface Type, part 1

Isotope	30 Year Decay			Sample Location ►	Moderator		Primary Coolant	Top Face External	
	Ci/gr fuel @ 30 yrs	Am-241 Ratio	Total Alpha Ration	Sample Type ►	Am	Total α	Total α	Am	Total α
Isotope	Ci/gr fuel @ 30 yrs	Am-241 Ratio	Total Alpha Ration	Sample Value (Ci/100 cm ²) ►	1.80 x10 ⁻⁶	7.20 x10 ⁻⁶	4.64 10 ⁻⁷	5.22 x10 ⁻⁸	7.50 x10 ⁻⁸
Am-241	2.63 x10 ⁻³	1.00 x10 ⁰	6.23 x10 ⁻¹	C C	1.8 x10 ⁻⁶	4.48421 x10 ⁻⁶	2.88983 x10 ⁻⁷	5.22 x10 ⁻⁸	4.67 x10 ⁻⁸
Np-237	2.26 x10 ⁻⁸	8.59 x10 ⁻⁶	5.35 x10 ⁻⁶	a o	1.55 x10 ⁻¹¹	3.85335 x10 ⁻¹¹	2.48327 x10 ⁻¹²	4.49 x10 ⁻¹³	4.01 x10 ⁻¹³
Pa-233	2.25 x10 ⁻⁸	8.56 x10 ⁻⁶	5.33 x10 ⁻⁶	l n	1.54 x10 ⁻¹¹	3.8363 x10 ⁻¹¹	2.47228 x10 ⁻¹²	4.47 x10 ⁻¹³	4 x10 ⁻¹³
Pa-234m	3.27 x10 ⁻⁷	1.24 x10 ⁻⁴	7.74 x10 ⁻⁵	c c	2.24 x10 ⁻¹⁰	5.57543 x10 ⁻¹⁰	3.59305 x10 ⁻¹¹	6.49 x10 ⁻¹²	5.81 x10 ⁻¹²
Pu-238	8.90 x10 ⁻⁵	3.38 x10 ⁻²	2.11 x10 ⁻²	u c	6.09 x10 ⁻⁸	1.51747 x10 ⁻⁷	9.77926 x10 ⁻⁹	1.77 x10 ⁻⁹	1.58 x10 ⁻⁹
Pu-239	1.02 x10 ⁻³	3.88 x10 ⁻¹	2.42 x10 ⁻¹	l n	6.98 x10 ⁻⁷	1.73912 x10 ⁻⁶	1.12077 x10 ⁻⁷	2.02 x10 ⁻⁸	1.81 x10 ⁻⁸
Pu-240	4.83 x10 ⁻⁴	1.84 x10 ⁻¹	1.14 x10 ⁻¹	a t ►	3.31 x10 ⁻⁷	8.23527 x10 ⁻⁷	5.30717 x10 ⁻⁸	9.59 x10 ⁻⁹	8.58 x10 ⁻⁹
Pu-241	8.76 x10 ⁻³	3.33 x10 ⁰	2.07 x10 ⁰	t r	6 x10 ⁻⁶	1.4936 x10 ⁻⁵	9.62543 x10 ⁻⁷	1.74 x10 ⁻⁷	1.56 x10 ⁻⁷
Pu-242	1.00 x10 ⁻⁷	3.80 x10 ⁻⁵	2.37 x10 ⁻⁵	e a	6.84 x10 ⁻¹¹	1.70502 x10 ⁻¹⁰	1.09879 x10 ⁻¹¹	1.98 x10 ⁻¹²	1.78 x10 ⁻¹²
Th-231	1.43 x10 ⁻⁸	5.44 x10 ⁻⁶	3.39 x10 ⁻⁶	d t	9.79 x10 ⁻¹²	2.43818 x10 ⁻¹¹	1.57127 x10 ⁻¹²	2.84 x10 ⁻¹³	2.54 x10 ⁻¹³
Th-234	3.27 x10 ⁻⁷	1.24 x10 ⁻⁴	7.74 x10 ⁻⁵	i	2.24 x10 ⁻¹⁰	5.57543 x10 ⁻¹⁰	3.59305 x10 ⁻¹¹	6.49 x10 ⁻¹²	5.81 x10 ⁻¹²
U-234	3.23 ⁻⁷	1.23 x10 ⁻⁴	7.65 x10 ⁻⁵	o	2.21 x10 ⁻¹⁰	5.50723 x10 ⁻¹⁰	3.5491 x10 ⁻¹¹	6.41 x10 ⁻¹²	5.74 x10 ⁻¹²
U-235	1.43 x10 ⁻⁸	5.44 x10 ⁻⁶	3.39 x10 ⁻⁶	n	9.79 x10 ⁻¹²	2.43818 x10 ⁻¹¹	1.57127 x10 ⁻¹²	2.84 x10 ⁻¹³	2.54 x10 ⁻¹³
U-236	5.02 x10 ⁻⁹	1.91 x10 ⁻⁶	1.19 x10 ⁻⁶	s	3.44 x10 ⁻¹²	8.55922 x10 ⁻¹²	5.51594 x10 ⁻¹³	9.96 x10 ⁻¹⁴	8.92 x10 ⁻¹⁴
U-238	3.27 x10 ⁻⁷	1.24 x10 ⁻⁴	7.74 x10 ⁻⁵		2.24 x10 ⁻¹⁰	5.57543 x10 ⁻¹⁰	3.59305 x10 ⁻¹¹	6.49 x10 ⁻¹²	5.81 x10 ⁻¹²

Table 3.0, Isotopic Calculations (Ci/100 cm³) by Surface Area, part 2

Isotope	30 Year Decay			Sample Location ▶ Sample Type ▶	LRS Floor		LRS Pipes External		Under Dump Tank	
	Ci/gr fuel @ 30 yrs	Am-241 Ratio	Total Alpha Ration		Am	Am	Am	Total α	Am	Total α
Am-241	2.63 x10 ⁻³	1.00 x10 ⁰	6.23 x10 ⁻¹	C C	4.98 x10 ⁻⁹	6.7 x10 ⁻⁹	1.13 x10 ⁻⁸		6.68 x10 ⁻⁷	4.67 x10 ⁻⁶
Np-237	2.26 x10 ⁻⁸	8.59 x10 ⁻⁶	5.35 x10 ⁻⁶	a o	4.27939 x10 ⁻¹⁴	5.76 x10 ⁻¹⁴	6.05 x10 ⁻¹⁴		5.74 x10 ⁻¹²	2.49933 x10 ⁻¹¹
Pa-233	2.25 x10 ⁻⁸	8.56 x10 ⁻⁶	5.33 x10 ⁻⁶	l n	4.26046 x10 ⁻¹⁴	5.73 x10 ⁻¹⁴	6.02 x10 ⁻¹⁴		5.71 x10 ⁻¹²	2.48827 x10 ⁻¹¹
Pa-234m	3.27 x10 ⁻⁷	1.24 x10 ⁻⁴	7.74 x10 ⁻⁵	c c	6.19186 x10 ⁻¹³	8.33 x10 ⁻¹³	8.75 x10 ⁻¹³		8.31 x10 ⁻¹¹	3.61628 x10 ⁻¹⁰
Pu-238	8.90 x10 ⁻⁵	3.38 x10 ⁻²	2.11 x10 ⁻²	u e	1.68525 x10 ⁻¹⁰	2.27 x10 ⁻¹⁰	2.38 x10 ⁻¹⁰		2.26 x10 ⁻⁸	9.84249 x10 ⁻⁸
Pu-239	1.02 x10 ⁻³	3.88 x10 ⁻¹	2.42 x10 ⁻¹	l n	1.93141 x10 ⁻⁹	2.6 x10 ⁻⁹	2.73 x10 ⁻⁹		2.59 x10 ⁻⁷	1.12802 x10 ⁻⁶
Pu-240	4.83 x10 ⁻⁴	1.84 x10 ⁻¹	1.14 x10 ⁻¹	a t ▶	9.1457810 ⁻¹⁰	1.23 x10 ⁻⁹	1.29 x10 ⁻⁹		1.23 x10 ⁻⁷	5.34149 x10 ⁻⁷
Pu-241	8.76 x10 ⁻³	3.33 x10 ⁰	2.07 x10 ⁰	t r	1.65874 x10 ⁻⁸	2.23 x10 ⁻⁸	2.34 x10 ⁻⁸		2.22 x10 ⁻⁶	9.68766 x10 ⁻⁶
Pu-242	1.00 x10 ⁻⁷	3.80 x10 ⁻⁵	2.37 x10 ⁻⁵	e a	1.89354 x10 ⁻¹³	2.55 x10 ⁻¹³	2.68 x10 ⁻¹³		2.54 x10 ⁻¹¹	1.1059 x10 ⁻¹⁰
Th-231	1.43 x10 ⁻⁸	5.44 x10 ⁻⁶	3.39 x10 ⁻⁶	d t	2.70776 x10 ⁻¹⁴	3.64 x10 ⁻¹⁴	3.83 x10 ⁻¹⁴		3.63 x10 ⁻¹²	1.58143 x10 ⁻¹¹
Th-234	3.27 x10 ⁻⁷	1.24 x10 ⁻⁴	7.74 x10 ⁻⁵	l	6.19186 x10 ⁻¹³	8.33 x10 ⁻¹³	8.75 x10 ⁻¹³		8.31 x10 ⁻¹¹	3.61628 x10 ⁻¹⁰
U-234	3.23 x10 ⁻⁷	1.23 x10 ⁻⁴	7.65 x10 ⁻⁵	o	6.11612 x10 ⁻¹³	8.23 x10 ⁻¹³	8.64 x10 ⁻¹³		8.2 x10 ⁻¹¹	3.57205 x10 ⁻¹⁰
U-235	1.43 x10 ⁻⁸	5.44 x10 ⁻⁶	3.39 x10 ⁻⁶	n	2.70776 x10 ⁻¹⁴	3.64 x10 ⁻¹⁴	3.83 x10 ⁻¹⁴		3.63 x10 ⁻¹²	1.58143 x10 ⁻¹¹
U-236	5.02 x10 ⁻⁹	1.91 x10 ⁻⁶	1.19 x10 ⁻⁶	s	9.50555 x10 ⁻¹⁵	1.28 x10 ⁻¹⁴	1.34 x10 ⁻¹⁴		1.28 x10 ⁻¹²	5.55161 x10 ⁻¹²
U-238	3.27 x10 ⁻⁷	1.24 x10 ⁻⁴	7.74 x10 ⁻⁵		6.19186 x10 ⁻¹³	8.33 x10 ⁻¹³	8.75 x10 ⁻¹³		8.31 x10 ⁻¹¹	3.61628 x10 ⁻¹⁰

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