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7. Abstract

Several small tanks under the management of Hanford Restoration Operations are investigated and relevant data are presented. The tanks' configuration and contents are listed. The tanks are 241-B-361, 241-T-361, 241-U-361, 241-Z-361, 241-Z-8, 241-CX-70, 241-CX-72, 270-E-1, and the 205-S process cell.

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EXECUTIVE SUMMARY

This report discusses eight radioactive waste storage tanks and one vault managed by Hanford Restoration Operations (HRO) under the Surplus Facilities Management Program (SFMP). These facilities range in size from 2,300 gallons to 40,000 gallons capacity. All have been out of service for at least fifteen years. The vault has been decommissioned and two tanks are currently undergoing decommissioning by removing their waste inventory. The remaining six tanks have been stabilized and are awaiting decommissioning as funding allows. The tanks and their contents are summarized in Table E-1.

Table E-1. Summary of Tank Contents.

TANK/VAULT	WASTE VOLUME	RADIOLOGICAL INVENTORY	HAZARDOUS
241-B-361	22,000 gal	2.4 kg Plutonium $2E^6$ Ci Strontium	None known
241-T-361	23,000 gal	2.6 kg Plutonium 7,800 Ci Cesium	None known
241-U-361	26,150 gal	69,000 kg Uranium 760 Ci Strontium 1,365 Ci Cesium	None known
241-Z-361	20,200 gal	26.8 kg Plutonium	None known
241-CX-70	750 gal	<1 g Plutonium	Chromium (EHW)
241-CX-72	1.6 m^3	0.2 kg Plutonium 6,000 Ci Cesium	None known
241-Z-8	500 gal	38 g Plutonium	None known
270-E-1	3,800 gal	Note 1	None known
205-S		Note 2	

[1] No sampling data is available; however, radiation measurements and process knowledge indicate that inventories are low.

[2] This vault has been decommissioned by removing its tanks and filling the vault with soil and concrete. The vault is covered with at least two feet of soil.

The data presented herein indicates that, with the exception of tank 241-T-361, no near-term actions are necessary for the continued safe storage of these facilities. Waste removal should proceed as soon as funding is available. Tank 241-T-361 should be opened to confirm the presence of 11,000 gallons of liquid waste. If present, the liquid should be removed.

SUMMARY OF RADIOACTIVE UNDERGROUND TANKS MANAGED BY HANFORD RESTORATION OPERATIONS

1. 241-B-361 SETTLING TANK: SUMMARY OF HISTORY AND CONTENTS

REFERENCE DRAWINGS: SK-2-4662 (BLUEPRINT/SKETCH)
H-2-44501 (PIPING/MAP)

The 241-B-361 settling tank is a cylindrical concrete structure located at coordinates N43400 and W52890*, approximately 900 ft northeast of the 221-B Building (Figures 1 and 2). It is 6 ft below grade. The tank was in service for two and a half years in the 1940s.

Constructed of unlined 6-in. reinforced concrete, the tank's capacity is 36,000 gal. It is 20 ft in diameter and 19 ft high with a domed top (Figure 3). Several risers are visible above grade. All eight risers were blanked during the B-231 isolation project in the early 1980s. The inlet and outlet lines on the east and west sides, respectively, are 3.5 in. (O.D.) carbon steel and have been blanked. Drawings indicate a 42-in. diameter manhole in the center of the tank.

The tank took in alkaline, low salt, low-level radioactive liquid waste from April 1945 to September 1947. The waste originated in the 224-B Building and in cells 5 and 6 of the 221-B canyon building. It was routed through the 241-B-154 diversion box before reaching tank 241-B-361, where it overflowed to the 216-B-5 reverse well.

A study of the 216-B-5 reverse well was conducted between 1979 and 1980 (Smith 1980). In the study, a number of monitoring wells were drilled, two of which were only a few feet away from tank 241-B-361. As neither of the monitoring wells showed any contamination, it was assumed that the tank was free of leaks.

During its period of service, 120 million gal of waste passed through the tank, containing 4.3 kg of plutonium (Smith 1980). Approximately half of this plutonium is estimated to have overflowed to the reverse well, leaving an estimated 2.4 kg remaining in tank 241-B-361. Quantities of strontium ($2E^6$ Ci) are also in the tank. The most recent sampling found was from April 30, 1979 (Horton 1979). The results can be found in Table 1.

There is some disagreement with respect to the quantity of waste in the tank. Although most reports estimate 32,000 gal (Smith 1980; Owens, et al. 1984; WHC 1987), another document (RHO 1985a), which details the pumping of the tank, indicates this value to be too high. According to this report, tank 241-B-361 was pumped from 9.85 ft to 9.46 ft. This corresponds to a present content of approximately 22,000 gal. Two reports (Smith 1980; WHC 1987) indicate that no pumpable liquids remain in the waste.

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* All coordinates referenced herein are per the Hanford Grid.

Figure 1. Reverse Well Location Map.

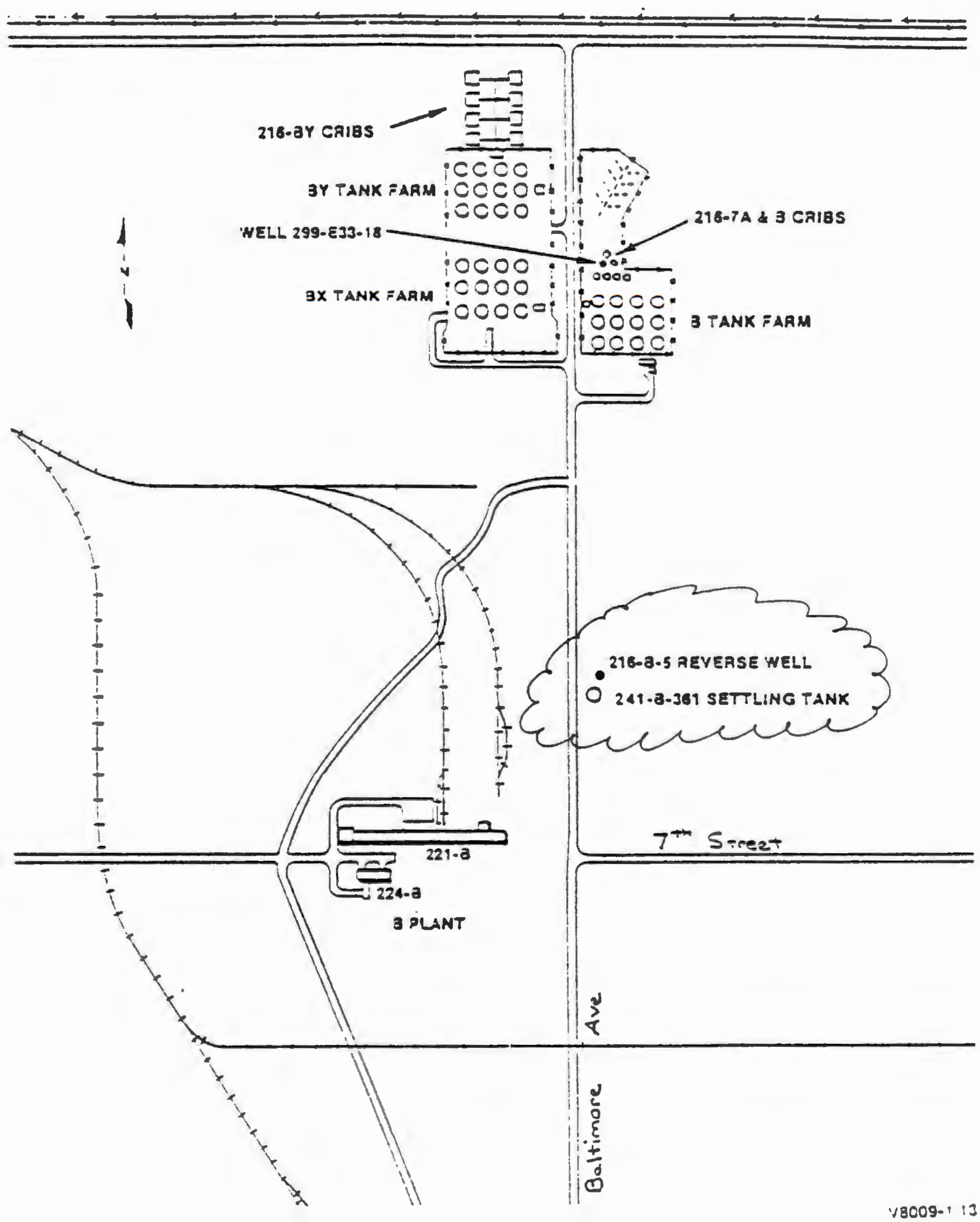

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Figure 2. 216-B-5 Reverse Well Disposal System.

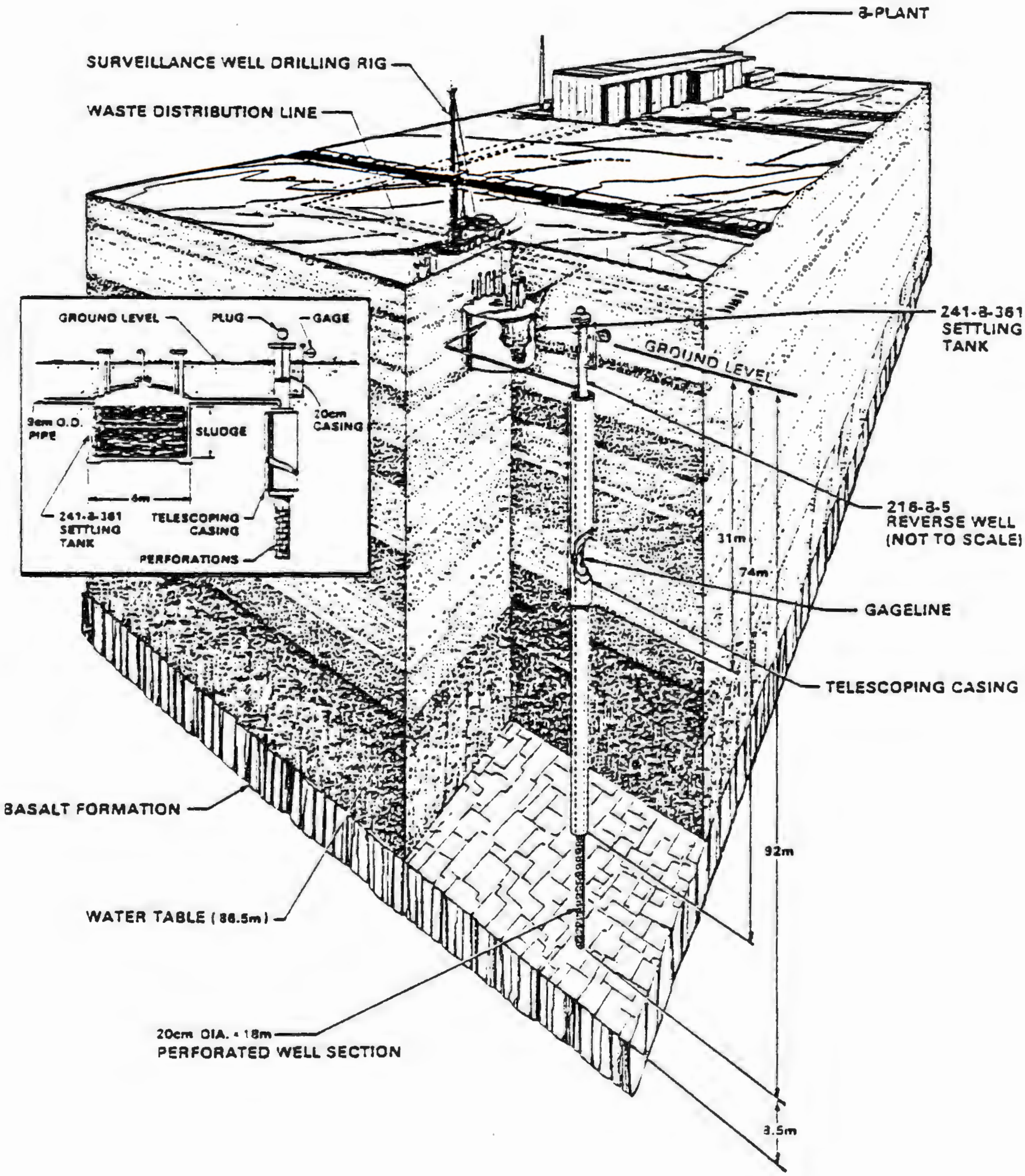

Figure 3. Settling Tank (241-B-361, 241-T-361, 241-U-361).

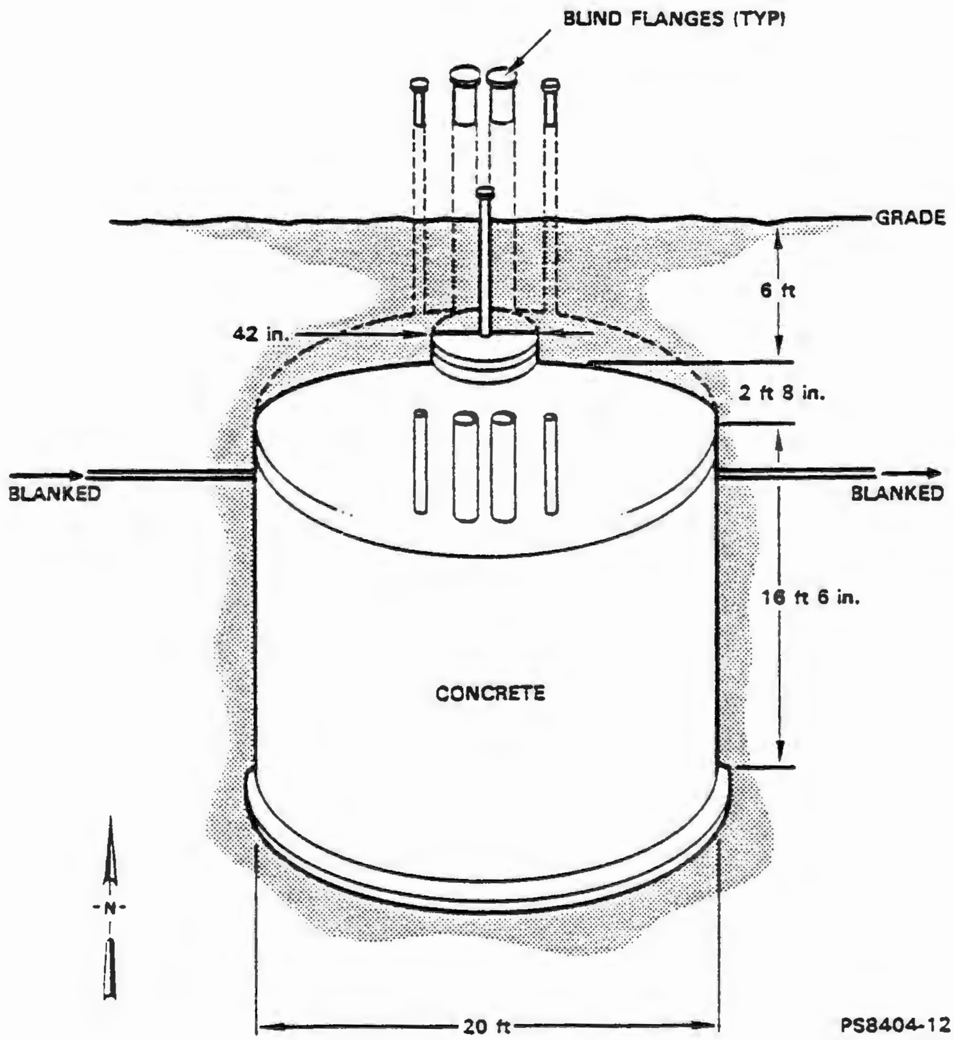

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Table 1. Analysis of 241-B-361 Settling Tank Sludge. (Smith 1980)

COMPONENT	SOLIDS, wt%	LIQUIDS
Al^{+3}	<0.06	---
Bi^{+2}	10.3	8.05 E^{-5} *M*
Fe^{+3}	1.3	2.0 E^{-4} *M*
F^{-}	0.04	1.0 E^{-2} *M*
La^{+3}	3.2	3.0 E^{-4} *M*
Mg^{+2}	0.5	<9.0 E^{-5} *M*
Mn^{+2}	3.0	<2.0 E^{-5} *M*
$NaAlO_2$	0.04	<4.05 E^{-4} *M*
Na_2CO_3	---	1.90 E^{-1} *M*
$NaNO_2$	---	3.0 E^{-2} *M*
$NaNO_3$	---	1.07 E^{+0} *M*
NaOH	---	2.4 E^{-1} *M*
Na_3PO_4	---	1.0 E^{-2} *M*
Na_2SO_4	---	4.0 E^{-2} *M*
Ni^{+2}	---	<5.2 E^{-5} *M*
NO_3^{-}	2.0	---
PO_4^{-3*}	3.4	---
SiO_4	0.4	2.0 E^{-3} *M*
SO_4^{-2}	0.2	---
^{239}Pu	3.4 μCi/g	6.1 E^{-7} μCi/mℓ
^{137}Cs	1.4 μCi/g	2.5 E^{-3} μCi/mℓ
$^{89-90}Sr$	23.0 μCi/g	3.1 E^{-5} μCi/mℓ
^{238}U*	1.1 E^{-5} g/g	8.4 E^{-6} g/mℓ

*All valences.

Note: No pumpable liquids contained in the tank.
Particle Density - 3.93 g/cm^3
Bulk Density - 1.29 g/cm^3
Moisture Content - ~72 wt%
Volume - 1.20 E^{+5} ℓ

The sludge is primarily bismuth phosphate (RHO 1982) and is described as being black with the consistency of pudding.

Another report (Harlow 1974) rated the heat generation of the tank as negligible. This is supported by a letter (Horton 1976a) which claims that the heat from strontium-89,-90 and cesium-137 is 0.00021 W/L. Based on this and the cost and risk involved in removing the sludge, Harlow's report recommends leaving the waste in the tank.

A more recent report (Owens, et al. 1984) describes plans to decommission the tank. It was originally planned to simply fill the retired, single-shell tanks at Hanford with gravel and stabilize them in place. This approach is inapplicable to tank 241-B-361, however, due to the large amount of waste it contains. The 30 percent void space in the gravel will not accommodate all of the sludge. Therefore, the report indicates that it will be necessary to remove the waste before proceeding with decommissioning. A specialized pump for viscous liquids would be required. Sluicing may also be necessary if the waste is too thick.

Since 1985 when tank 241-B-361 was interim stabilized, no surveillance data has been compiled. The Hanford Surplus Facilities Program Plan (Hughes, et al. 1989) shows decommissioning of this tank in the year 2003.

2. 241-T-361 SETTLING TANK: SUMMARY OF HISTORY AND CONTENTS

REFERENCE DRAWINGS: SK-2-4661 (TOP VIEW SKETCH)
SK-2-4662 (SKETCH OF IDENTICAL 241-B-361)
H-2-71641 (AREA MAP)

The 241-T-361 settling tank is a cylindrical concrete structure located at coordinates N43225 and W74000, approximately 700 ft southwest of the 224-T Building (Figure 4). It is 6 ft below grade. The tank was in service for slightly over a year in the 1940s.

Constructed of unlined 6-in. reinforced concrete, the tank's capacity is 36,000 gal. It is 20 ft in diameter and 19 ft high with a domed top (see Figure 3). Several risers are visible above grade. All eight risers were blanked during the B-231 isolation project. The inlet and outlet lines on the south and north sides, respectively, are 3.5 in. (O.D.) carbon steel and have been blanked. Drawings indicate a 42-in. diameter manhole in the center of the tank. The structural condition of the tank is unknown, although there is no indication that the tank has deteriorated.

Tank 241-T-361 took in waste from cells 5 and 6 in the 224-T Building from August 1946 to October 1947. Overflow was sent to the 216-T-6 crib.

Records indicating the type and amount of waste in the tank are incomplete and often contradict one another, especially with respect to the amount of plutonium contained. Several reports (Owens, et al. 1984; WHC 1987) estimate that the tank contains 15,500 Ci of beta/gamma and 2 kg of plutonium. However, a report (Harlow 1974) indicates that 170 million gal of waste passed through the tank containing an average of 2.29 E^{-6} g/gal of plutonium and 1.33 E^{-4} g/gal of uranium. Assuming no other waste was added to the tank, this would mean that the tank could contain no more than 390 g of plutonium. Sampling of tank 241-T-361 was done in October 1976 (Horton 1977). Results can be found in Table 2. These sampling results indicate an inventory of 2,600 grams of plutonium and 7,800 curies of cesium. No records indicate that anything has been added to the tank since it was used in the 1940s.

In 1974, 5.21 ft of sludge was covered with 7.37 ft of a yellow liquid, giving a total waste level of 12.58 ft. In 1985, the tank was pumped to 10.04 ft (RHO 1985b; RHO 1985c). Assuming that the sludge layer remains relatively unchanged and only liquid was removed, the pumping would have left a content of approximately 12,000 gal of sludge and 11,000 gal of liquid, for a total of 23,000 gal of waste.

Another report (Harlow 1974) rated the heat generation of the tank as negligible. This is supported by a letter (Horton 1977) which claims that the heat from strontium-89,-90 and cesium-137 is 0.000818 W/L. Based on this and the cost and risk involved in removing the sludge, Harlow's report recommends leaving the waste in the tank. Harlow cited no other radiological factors or chemical hazards to support this conclusion.

Figure 4. Separations Area -- 200 West.

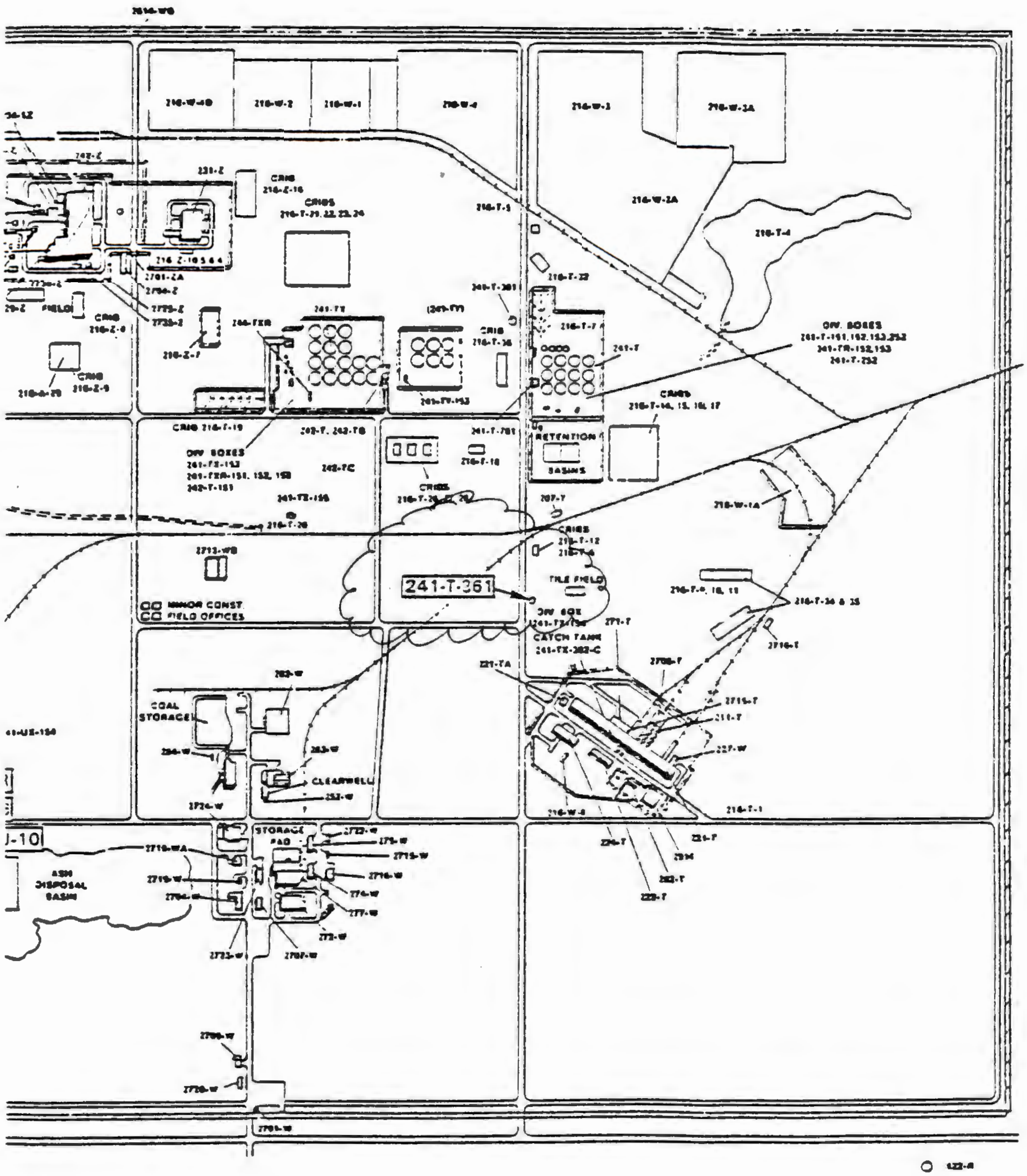

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Table 2. Analyses of Tank 241-T-361 Sludge.

Bulk Density	2.53 g/cc
Particle Density	3.91 g/cc
H_2O	61.3%
Al	<1.0%
Fe	2.0%
CO_3	0.9%
Ca	0.6%
NO_2	1.4%
NO_3	17.4%
SO_4	<1.0%
PO_4	1.0%
Ni	5.8%
Si	1.8%
Na	3.1%
Mg	0.2%
Mn	1.7%
Pu	2.30×10^{-5} g/g
$^{89\,90}Sr$	0.120 μCi/g
^{137}Cs	67.6 μCi/g

A more recent report (Owens, et al. 1984) describes plans to decommission the tank. It was originally planned to simply fill the retired, single-shell tanks at Hanford with gravel and stabilize them in place. This approach may not be applicable to tank 241-T-361, however, due to the significant amount of waste it contains. The 30 percent void space in the gravel may not accommodate all of the sludge. Therefore, the report claims that it could be necessary to remove the waste before proceeding with decommissioning. A specialized pump for viscous liquids would be required. Sluicing may also be necessary if the waste is too thick.

Since 1985 when tank 241-T-361 was isolated, no surveillance data has been compiled. The Hanford Surplus Facilities Program Plan (Hughes, et al. 1989) for fiscal year 1990 shows this tank to be scheduled for decommissioning in the year 2003.

3. 241-U-361 SETTLING TANK: SUMMARY OF HISTORY AND CONTENTS

REFERENCE DRAWINGS: SK-2-4661 (TOP VIEW SKETCH)
SK-2-4662 (SKETCH OF IDENTICAL 241-B-361)
H-2-44511 SHEET 61 (AREA MAP)

The 241-U-361 settling tank is a cylindrical concrete structure located at coordinates N37830 and W74160, approximately 800 ft west of the 221-U Building (Figure 5). It is 6 ft below grade. The tank was in service from November 1951 to June 1967.

Constructed of unlined 6-in. reinforced concrete, the tank's capacity is 36,000 gal. It is 20 ft in diameter and 19 ft high with a domed top (see Figure 3). Several risers are visible above grade. All eight risers were blanked during the B-231 isolation project. The inlet and outlet lines on the south and north sides, respectively, are 3.5 in. (O.D.) carbon steel and have been blanked. Drawings indicate a 42-in. diameter manhole in the center of the tank. The structural condition of the tank is unknown, although there is no indication that the tank has deteriorated.

Tank 241-U-361 took in low-level waste from the uranium recovery process in the 221-U Building and decontamination wastes from the 224-U Building. The tank overflowed to cribs 216-U-1 and 216-U-2.

Sixty million gallons of waste passed through the tank, containing an average of 7.08 E^{-7} g/gal of plutonium and 6.71 E^{-2} g/gal of uranium. These values calculate to a total of 42.5 g of plutonium and 4,026 kg of uranium, but it is unknown how much of these elements remain in the tank. The only sampling data found on file was from April 1976 (Table 3) (Horton 1976b). Based on this sampling data, it is estimated that the tank contains 760 Ci of strontium-90, 1365 Ci of cesium-137, 69,000 kg of uranium, and less than one gram of plutonium.

A report (Owens, et al. 1984) indicates 27,500 gal of waste in the tank, and another document (RHO 1985d) details a pumping that decreased the liquid content in the tank by approximately 1350 gal. This decreased the waste level to 11.8 ft. Little or no liquid is reported to be in the tank, and the sludge is described as having the consistency of soft mud.

Another report (Harlow 1974) rated the heat generation of the tank as negligible. Based on this and the cost/risk involved in removing the sludge, Harlow's report recommends leaving the waste in the tank. Harlow cited no other radiological factors or chemical hazards to support this conclusion.

A more recent report (Owens, et al. 1984) describes plans to decommission the tank. It was originally planned to simply fill the retired, single-shell tanks at Hanford with gravel and stabilize them in place. This approach is inapplicable to tank 241-U-361, however, due to the large amount of waste it contains. The 30 percent void space in the gravel will not accommodate all of the sludge. Therefore, the report claims that it will be necessary to remove the waste before proceeding with decommissioning. A specialized pump for viscous liquids would be required. Sluicing may also be necessary if the waste is too thick.

Figure 5. Location Map of 241-U-361.

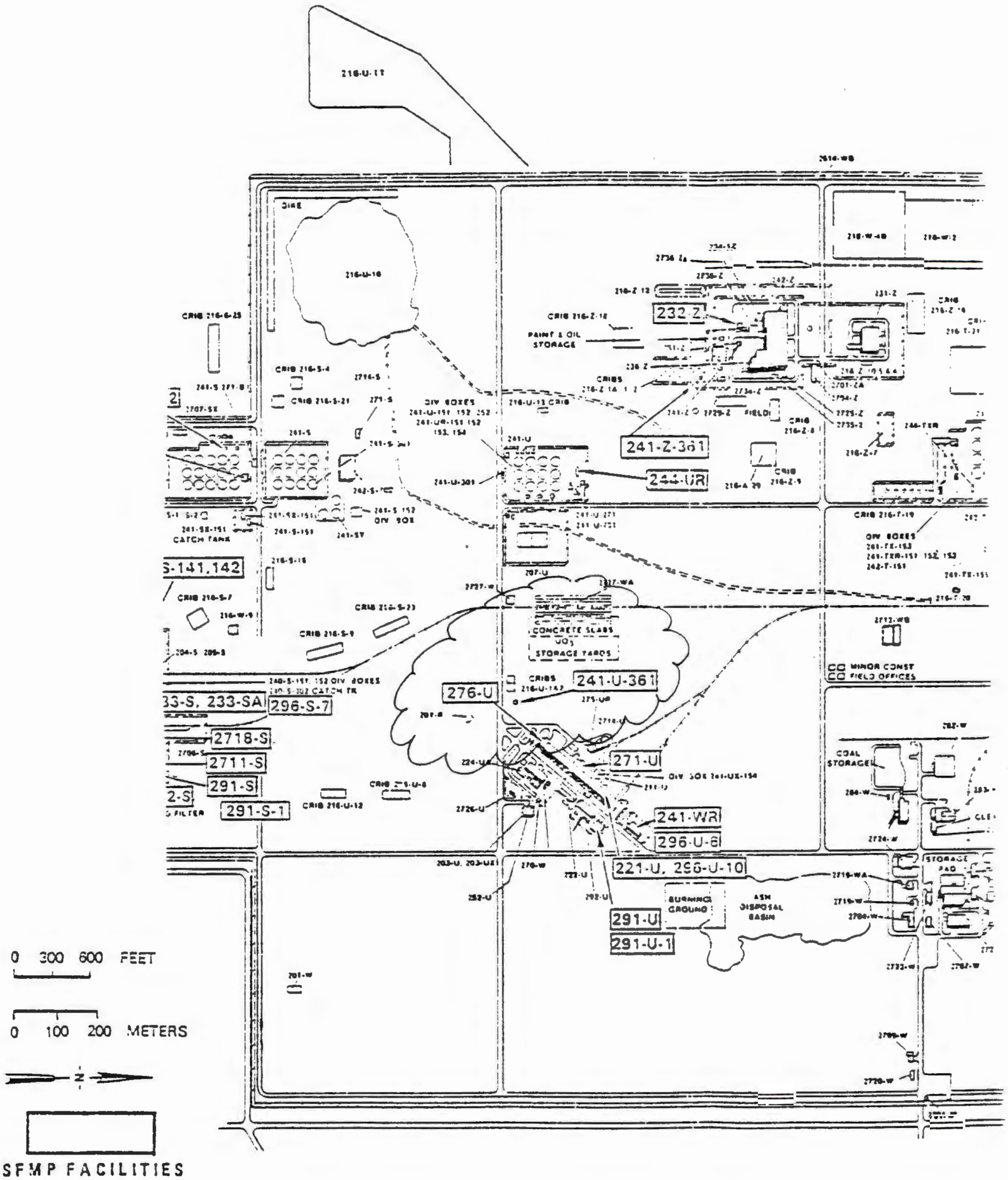

Table 3. Analysis of Tank 241-U-361 Sludge.

Bulk Density	1.49 g/cc
Particle Density	5.97 g/cc
H_2O	65.6%
Al_2O_3	2.4%
Na_2CO_3	<1.0%
FeOH	2.9%
$NaNO_2$	<1.0%
$NaNO_3$	27.2%
Mg	0.06%
Mn	0.6%
Na_2SO_4	1.3%
Na_3PO_4	<1.0%
Ni	0.5%
SiO_2	0.3%
Na	4.4%
U	0.133 μCi/g
Pu	9.97×10^{-7} μCi/g
$^{89+90}Sr$	4.9 μCi/g
^{137}Cs	8.8 μCi/g

Since 1985 when tank 241-U-361 was interim stabilized, no surveillance data has been compiled. The Hanford Surplus Facilities Program Plan (Hughes, et al. 1989) for fiscal year 1990 shows this tank scheduled for decommissioning in the year 2003.

4. 241-Z-361 SETTLING TANK: SUMMARY OF HISTORY AND CONTENTS

REFERENCE DRAWINGS: H-2-16460 (BLUEPRINT)
H-2-44511 SHEET 87 (PIPING/MAP)

The 241-Z-361 settling tank is a rectangular concrete structure located at coordinates N39500 and W76600, approximately 350 ft south of the 234-5Z Building (Figure 6). It is 2 ft below grade. Put into service in 1949, the tank took in neutralized, low-salt, aqueous waste from Z-Plant until the spring of 1973.

The structure's inner dimensions are 26 ft by 13 ft, with the inner height varying from 17 to 18 ft due to a sloping bottom (Figures 7 and 8). Total capacity of the tank is slightly less than 40,500 gal. The walls of the tank are 12-in. thick concrete lined with 3/8-in. steel. According to the blueprints, the top of the tank contains several risers, a 3-ft diameter manhole cover at the north and south ends, and a central concrete plug 4 ft in diameter. However, according to a memo (Cowley 1975), the central entrance could not be found, and a 3-in. concrete slab was poured over much of the tank. Later, a survey by the U.S. Energy Research and Development Administration (Burton 1975) reported that the central manhole had been core drilled for sampling purposes. This indicates that the plug had eventually been found, but whether or not it is currently an accessible entrance remains unknown. The two inlets on the north end and the outlet on the south end are blanked. The risers are reported as being covered with plastic and taped (RHO 1982).

Z-Plant has had several operating modes, such as:

- Oxalate precipitation and filtration
- Plutonium reduction
- Recuplex (1956 to 1962)
- Weapons part fabrication (1949 to 1965)
- Oxalate filtrate ion exchange (1962 to 1964)
- Incinerator (1962 to 1973)

These and other operations produced waste streams (usually flushed in 3000 L batches) from:

- The Hydro-Flurinator off-gas water jet
- Cooling waters and condensates
- Lab wastes and hood floor drains
- The off-gas scrubber of the Z-Plant incinerator (the wastes from the incinerator contained fly ash, which is thought to have caused the dark color of the upper portion of the tank's sludge)

Figure 6. Major Z-Plant Facilities.

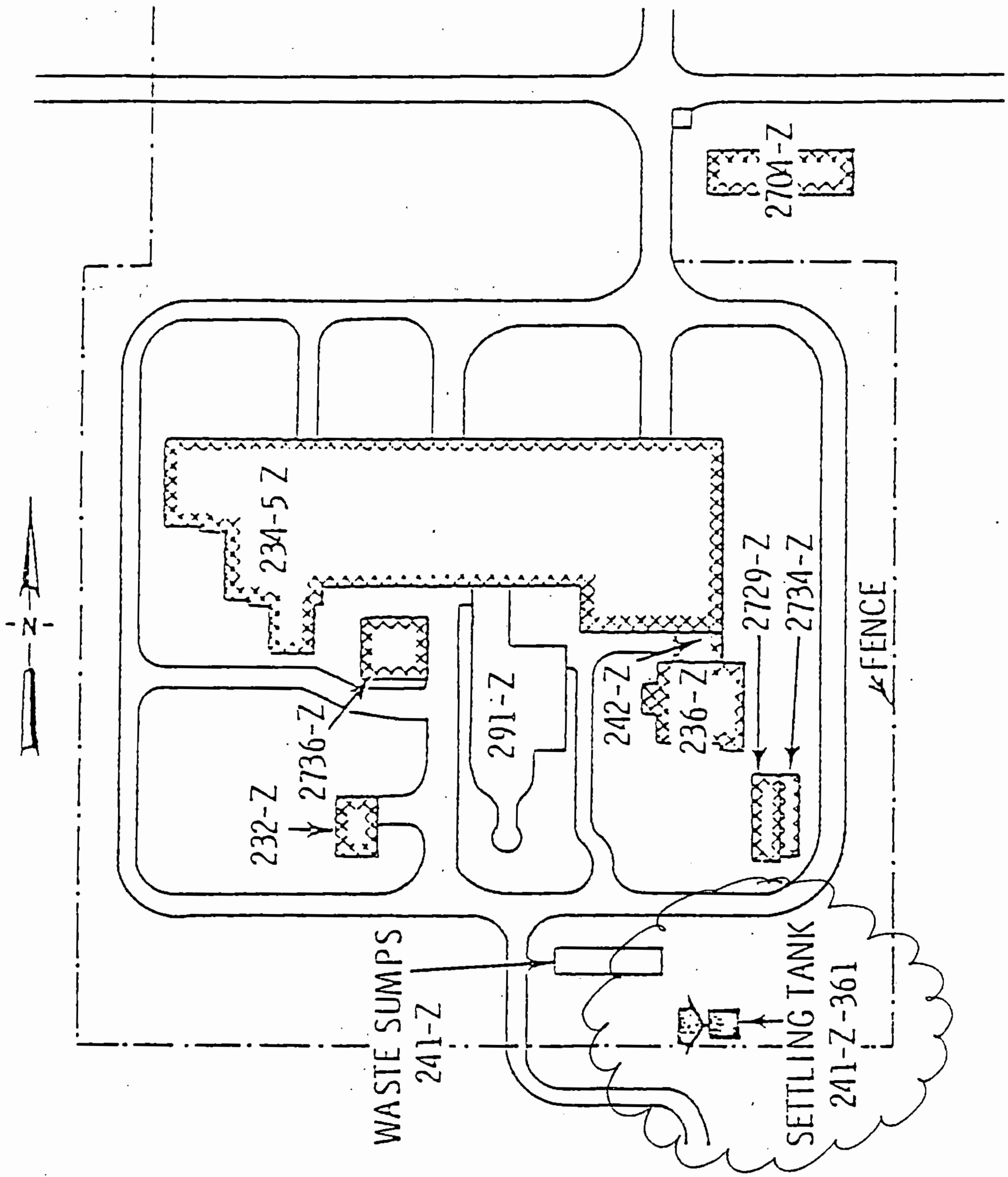

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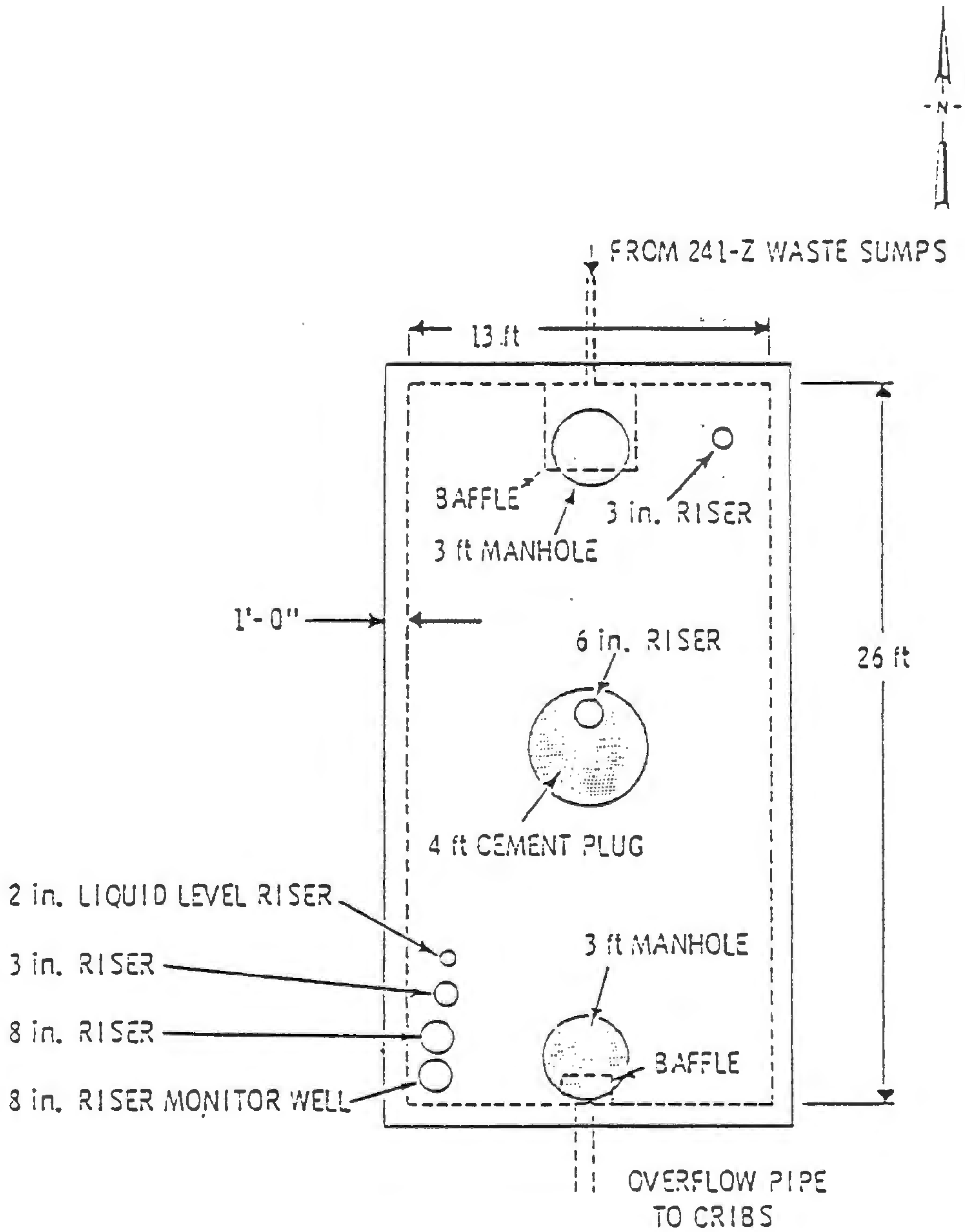
Figure 7. Plan View of 241-Z-361



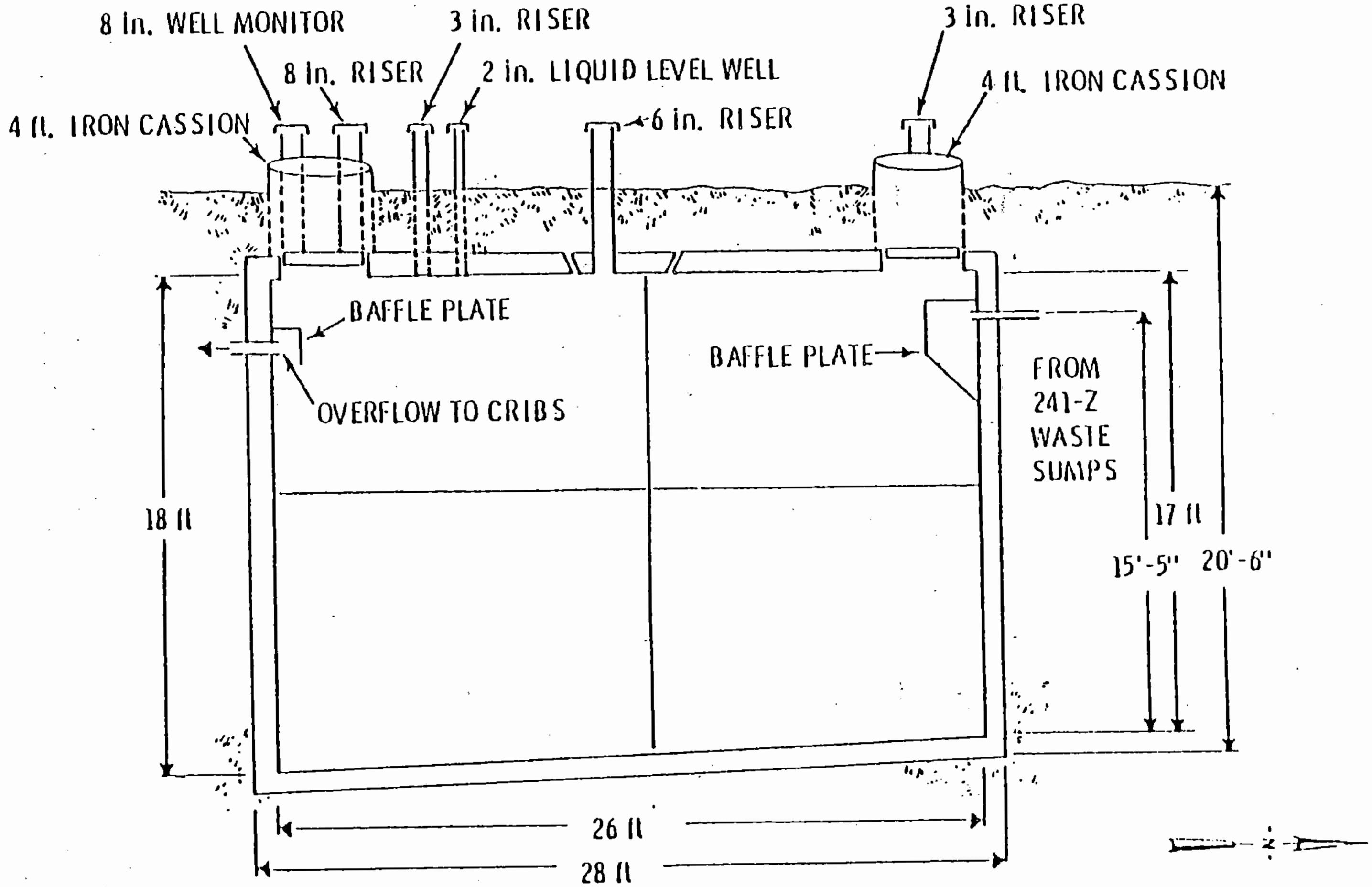


Figure 8. Elevation View of 241-Z-361 Settling Tank.

These waste streams were routed through the 241-Z sump tanks for neutralization, sent to tank 241-Z-361 to settle out any solids, and overflowed into four cribs:

- Z-1 and Z-2 from June 1949 to June 1952
- Z-3 from July 1952 to May 1973
- Z-12 from March 1959 to May 1973

The tank went out of use in May 1973, at which time all the lines were blanked, and the tank's liquid level was left at the overflow point.

In May 1975, approximately 21,000 gal of supernate was pumped from one of the 8-in. risers and trucked to tank farms. This left only about 200 gal of standing liquid and approximately 8 ft, or 20,000 gal of wet, brown-black sludge. The sludge is described as having the consistency of mud or pudding and is too viscous to be pumped by conventional means as is.

Due to the different modes of operation at Z-Plant, slightly different wastes were produced. Samples have shown that these wastes remain layered in the tank, indicating that the contents have mixed only slightly.

In the earlier years of Z-Plant operation, the waste was treated with soda ash. Evidence of this is given by the white streaks in the lower, brownish sludge layers of the tank. The earlier Z-Plant operations also produced less plutonium, as did the operations near the end of the tank's history (Davenport, et al. 1977). Because of this, plutonium concentrations are probably slightly greater in the middle layers of the sludge. Due to the layered, nonhomogeneous nature of the sludge, it has been difficult to get samples that are considered representative of the entire tank.

The last sampling found on file was from August 1978 (Bruns 1978). The samples were analyzed using both chemical methods and a foil study. They were taken through four different dry wells installed earlier that year. The wells were four inch aluminum pipe inserted through the two manhole covers and two of the risers. Moisture content in the tank was found to be 30 percent plus or minus 5 and the plutonium concentrations are shown in Table 4. Other radioactive products were not mentioned and the tank has a low enough dose rate to suggest that these do not present a hazard, but further testing may be required to confirm this. Records on six prior samplings were found on file, but none appeared more accurate or seemed more complete than the one included here.

A report (Dodd 1975) estimated a plutonium content from 0.4 to 1.1 g/L of wet sludge. This corresponds to 1.0 to 3.1 g/L of air dried solids, and means that the total quantity of plutonium in the tank is between 30 and 75 kg. According to a letter (Roecker 1977), the lower values in this range seem to have more credibility. This is supported by the included sample in Table 4 (Bruns 1978).

Neglecting considerations of neutron absorbers, even the highest estimates of plutonium content would be subcritical with up to 85 percent moisture loss, according to a letter (Carter, et al. 1976). When the

Table 4. Foil Versus Core Analysis in Tank 241-Z-361 - Plutonium Concentrations.

FT FROM BOTTOM OF TANK	WELL #1 (g/l)		WELL #2 (g/l)		WELL #3 (g/l)		WELL #4 (g/l)	
	CHEM	FOIL	CHEM	FOIL	CHEM	FOIL	CHEM	FOIL
1	0.41	0.12	0.16	0.14	0.21	0.12	0.64	0.10
3	0.56	0.48	0.19	0.38	0.28	0.36	0.81	0.18
5	0.79	0.33	0.26	0.26	0.41	0.29	1.27	0.24
7	0.54	0.28	0.10	0.17	0.17	0.42	0.40	0.27

quantities of iron, sodium, and cadmium in the tank are considered, the contents should be safely subcritical assuming loss of liquid by tank leakage plus drying out.

When the tank was pumped in 1975, it was intended that the solids be left in place and the vessel be filled with gravel. However, this approach was discovered to be inapplicable to tank 241-Z-361. The gravel's 30 percent void space will not accommodate the quantity of waste in the tank.

According to a letter (Felt 1976), other options were available. Drumming the waste or processing it for plutonium recovery are the two strongest possibilities. Both of these alternatives would first require drying of the waste. Due to the possibility of creating a critical system by mixing, however small the chance may be, drying the waste in place may be the best approach. A letter (Crawley 1976b) briefly outlines a plan to evaporate the moisture from the sludge. The evaporation would take 3 to 6 months and involved providing an air flow through the tank at a rate of approximately 25 to 50 cfm and a source of heat provided by 1000 W heat lamps. However, no documentation was found that suggests that this evaporation ever took place.

The option of sending the sludge to tank farms was considered as an option only in the case of a leak or other emergency, due to plans at that time to avoid adding actinide waste to the tanks. Two backup options were suggested in tank farms (Felt 1976). The first was to slurry and pump the waste into trucks for transport to tank D-9. This plan required new piping to tank 241-Z-361 and a protective containment structure over tank D-9. The second plan was to slurry the waste and transfer it to tank D-5 were it could be pumped directly to the 242-T evaporator.

A criticality incident alarm was installed in late 1975 and indicated a gamma reading of 0.4 mrem/hour and a neutron reading of 0.1 mrem/hour. The tank's contents have remained relatively unchanged, and the radiation levels are expected to remain the same as long as this is the case. In a letter (Crawley 1976a), the safety limit for neutron and gamma radiation was estimated to be at 1 mrem/hour. A Criticality Prevention Specification (WHC 1991) that includes this tank indicates that a criticality incident is not possible under all credible scenarios of surveillance. Any additional activities must be evaluated separately.

Hanford Restoration Operations has compiled no surveillance data on tank 241-Z-361 and there are no immediate plans for decommissioning.

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5. 241-CX-70 TANK: SUMMARY OF HISTORY, CONTENTS, AND DECOMMISSIONING

REFERENCE DRAWINGS: H-2-4319 (BLUEPRINT)
SK-2-56955 (PIPING/MAP)

The Strontium Semiworks Facility was a pilot plant that tested separation processes. It was used to evaluate operations such as the reduction-oxidation (REDOX) and the plutonium/uranium extraction (PUREX) plants. The plant was also used for strontium recovery and for brief testing of americium and curium isotope recovery.

The 241-CX-70 tank took in high level process waste from the REDOX process at the Semiworks Facility. It is a cylindrical, concrete structure located at coordinates N42100 and W50200, or 126.6 ft south southeast of the 201-C building site (Figure 9). It is 11 ft below grade. Built in 1952, it was used until 1957. Procedures for the Strontium Semiworks suggest that tank 241-CX-70 was a holding unit for waste before it was sent to nearby Tank Farm (WHC 1990). No data has been found pertaining to the quantity of material that passed through the tank.

The structure's inner dimensions are 20 ft in diameter and 15 ft high, with a capacity of 30,000 gal (Figure 10). The sides and top are constructed of 12-in. concrete. The concrete at the bottom varies in thickness from 2 ft at the edges to 9 in. at the center. The tank is lined with 1/4-in. stainless steel. Two 4-in. fill pipes enter from the north side and nine risers extend above ground level. The fill pipes have been blanked. There is a 42-in. manhole with a concrete cover in the northeast section of the tank.

In 1979, the liquid inventory remaining from the 1950s was pumped out, leaving 10,300 gal of very soft sludge. The supernate was pumped through temporary overground lines to the C-154 diversion box, then through an existing underground network to the 244-CR vault, were jumper connections routed the waste to tank 011-CR.

After several years, 241-CX-70 decommissioning activities were initiated as part of the Strontium Semiworks decommissioning project, which began in 1984. In the summer of 1987, plans were made to remove the waste remaining in tank 241-CX-70 as far as practical. Again the waste would be pumped via the 244-CR vault to a double-shell tank in 200 East Area Tank Farms. The sluicing and pumping wound up being done in three sessions in 1988 (Cummings, et al. 1988). The first attempt was discontinued due to the inefficiency of the 360-degree tank sluicer. Forty thousand gallons of water was used to remove only 770 gal of waste. In March 1988, after several repairs and the replacement of the 360-degree sluicer with a fire hose nozzle, the project was restarted and operated more efficiently. The system was shut down only for basic modifications, and the third sluicing session continued until there was not enough waste in the tank to make sluicing efficient. A total of approximately 10,000 gal of waste was removed using 140,000 gal of water. This left approximately 500 gal of liquid and 250 gal of solid that remain today.

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Figure 9. Site Facilities Map of Strontium Semiworks.

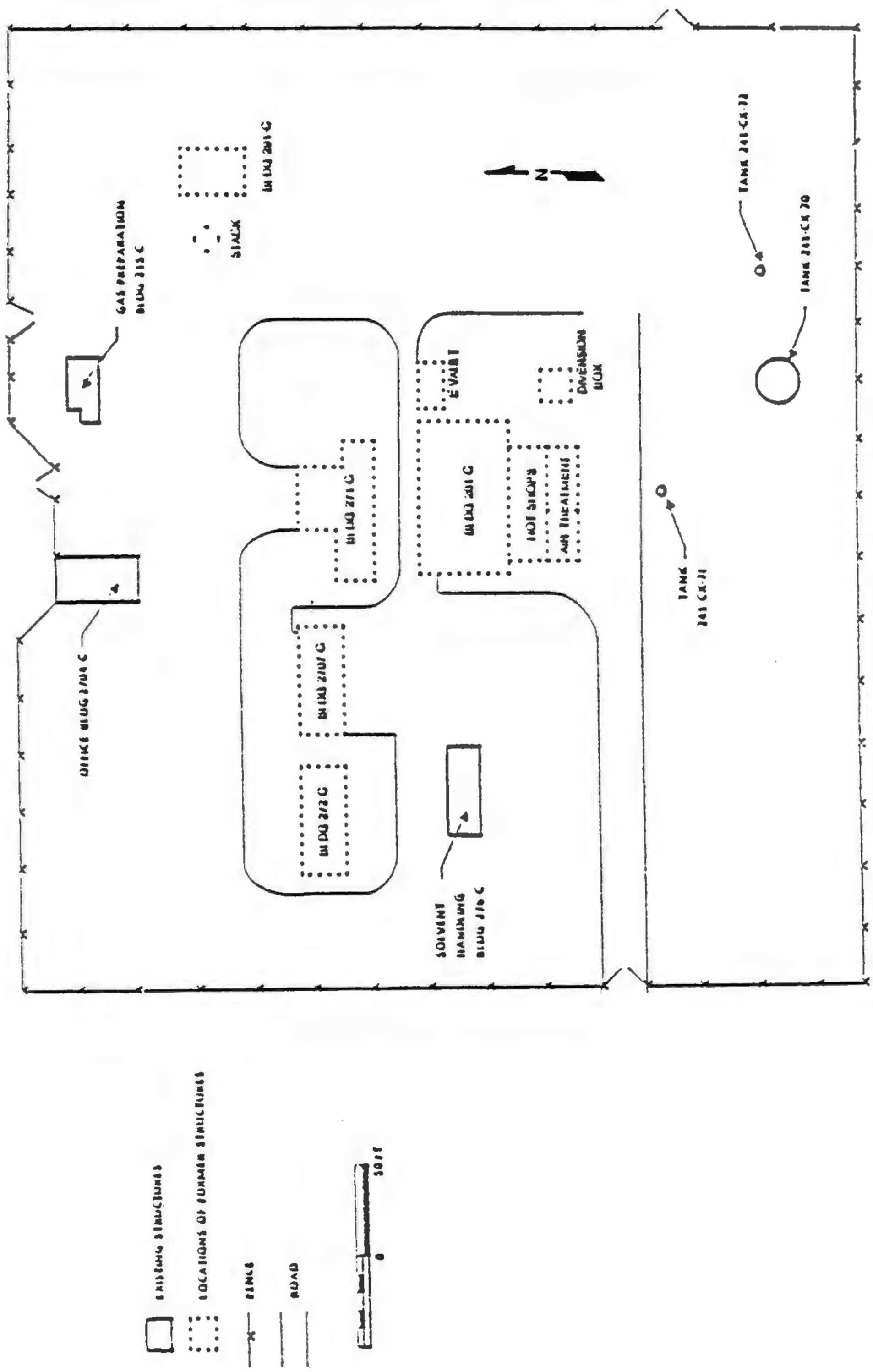
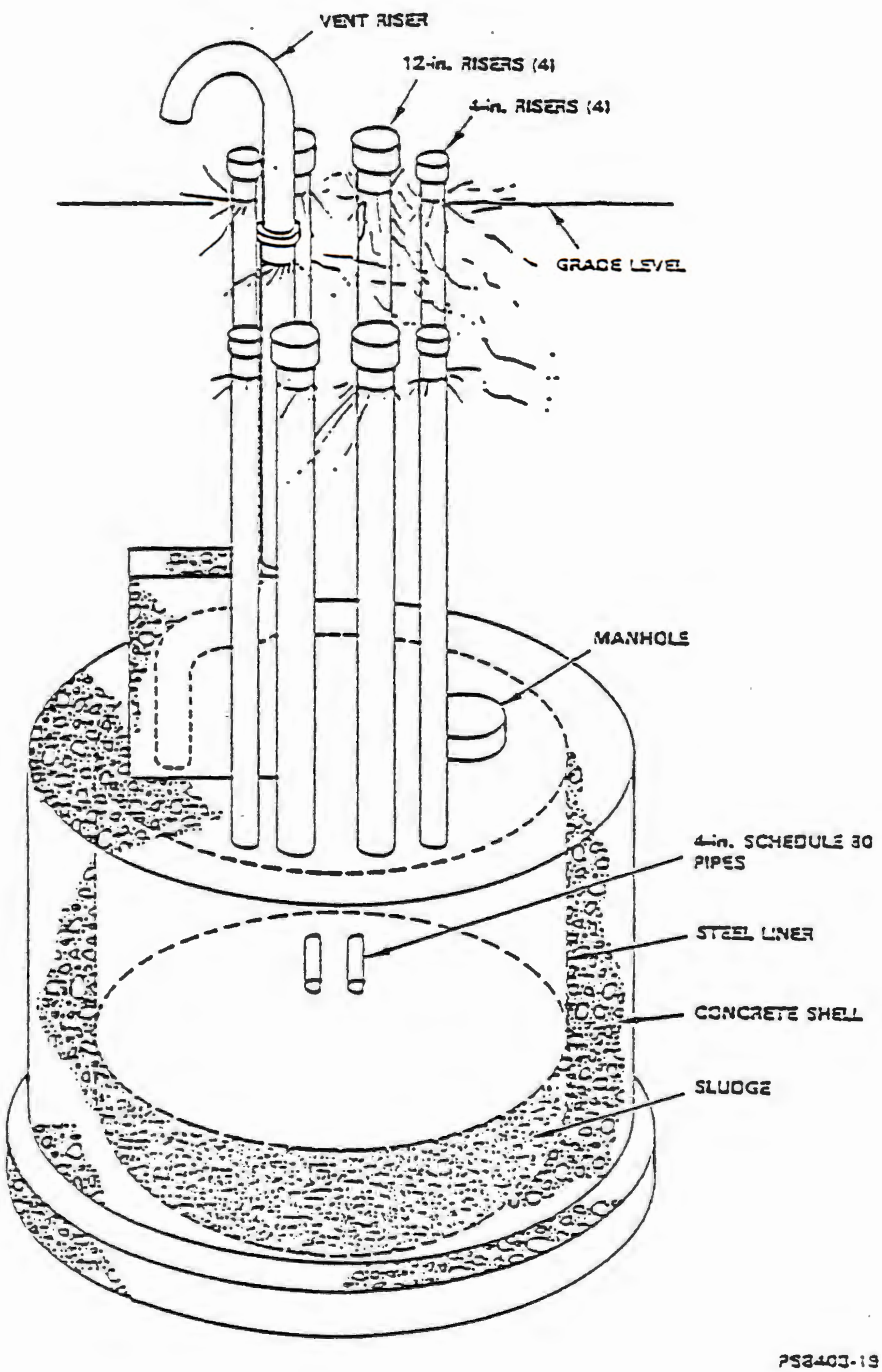


Figure 10. Tank 241-CX-70 Configuration Prior to Start of Decommissioning.



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A grab sample was obtained prior to sluicing, using a bottle and string. Due to the method used and the two phases present in the tank, the accuracy of these samples is questionable.

Samples were also taken periodically during each of the sluicing sessions through a transfer line. The results could then be adjusted to calculate the amount of contaminants in the waste before they were diluted with water.

Halogenated hydrocarbons were found to comprise 0.0009 wt% of the tank's contents. Therefore the tank is undesignated for organic waste, as it falls well below the 0.02 wt% required for a halogenated hydrocarbon designation.

The concentrations of inorganic compounds were calculated from the concentration of cations and anions found in the waste. These calculated values are:

- sodium nitrite — Equivalent Concentration = 0.0232%
- sodium fluoride — Equivalent Concentration = 0.00234%
- sodium nitrate — Equivalent Concentration = 0.00163%
- sodium chromate — Equivalent Concentration = 0.00011%
- aluminum sulfate — Equivalent Concentration = 0.00039%

The total is 0.028 percent, which qualified as extremely hazardous waste (EHW). In addition, sufficient quantities of chromium were present to designate the waste as EHW.

During sluicing, caustic was periodically added to meet the Tank Farm requirement of transferring waste at a pH of 11.0 or greater. Therefore, after sluicing operations the tank had a pH of 13.1. According to WAC-173-303-090, any waste with a pH greater than 12.5 due to sodium hydroxide content is designated D002 for a corrosive waste.

With respect to the pH problem, a letter from Decommissioning Engineering (Speer 1991a) indicates that sulfuric acid has been added to the tank, lowering the pH to 10.5. In view of this, it is assumed that the waste is no longer considered corrosive.

The tank has a transuranic (TRU) content of 390-690 nCi/g of solid, which is high enough to warrant further decommissioning. Quantities of cesium-137 and strontium-90 were also found in the waste. Most of the fission products are located in the solid. The most recent data found on the contents of the tank are three grab samples taken August 17, 1988 (Smoot 1988), which is after the last sluicing. The results can be found in Table 5.

Procedures and equipment are in place for the final removal of tank waste and this operation is underway. According to WHC-SD-DD-SAD-001 (WHC 1990), the end point for the decommissioning will be when the pH has been adjusted to less than 12.5, the liquid has been pumped or absorbed, and "all solid waste that is not adhered to the walls has been removed." However, document SD-DD-TI-034 (Cummings, et al. 1988) adds that if complete sludge removal is not possible, then it should be removed far enough such that the hazardous waste is gone and the TRU content is below 100 nCi/g. As was

Table 5. Material Inventory. (WHC 1990)

MATERIAL INVENTORY[1]
TANKS CX-70, CX-71, CX-72
STRONTIUM SEMIWORKS

	241-CX-70	241-CX-71	241-CX-72
Contents	2/3 liquid; 1/3 solid	1/2 liquid; 1/2 solid Limestone Filtration Media (calcium carbonate)	Dry Sludge
	Liquid[2] - Total beta = 34-315 nCi/g Cs-137 = 4-235 nCi/g Sr-90 = negligible[3]	Liquid - Sr-90 = 4 nCi/g Cs-137 = 2 nCi/g	Cs-137 - 2133 microCi/g
	Solid - Total beta = 76,000-153,000 nCi/g Cs-137 = 760-1,530 nCi/g[3] Sr-90 = 19,000-38,000 nCi/g[3]	Liquid plus limestone media Sr-90 = 3 nCi/g Cs-137 = 1 nCi/g	
TRU	Liquid - 2 nCi/g Solid - 690 nCi/g	Liquid plus solid: 2-8 nCi/g[5]	Solid: 23,513 nCi/g[6] (Pu) 2,613 nCi/g[6] (Pu)
Quantity	2850 liters[4]	4 cubic meters	1.6 cubic meters
Source	Sampling & Analyses	Historical Data	Radiation Measurements Historical Data Calculation
References	(Herting 1988; WHC 1988)	(WHC 1989c, 1990f)	(Subrahmanyam 1989; WHC 1989a, 1990b)

NOTES

1. Fractional values rounded upward to integer values.
2. Concentration units converted from microcuries per liter to nanocuries per gram based on a liquid bulk density of 1.04 gram per milliliter.
3. Strontium concentrations based on observation that strontium activity historically represents 10% and 25% of total beta activity for liquid and solid, respectively. Cesium concentrations in solid based on observation that cesium activity represent 1% of total beta activity for solid.
4. Waste volume will increase due to neutralization requirements.
5. Maximum TRU concentration based on assumption that all plutonium (as Pu-239) flushed through system was retained in liquid & limestone media.
6. Estimates of TRU are reported for weapon grade and 9% reactor-fuel grade plutonium. The upper bound estimate (weapons grade) is used for criticality determinations.

mentioned, the first step of lowering the pH has already been completed. Safety evaluations based on possible accident analysis and the contaminants in the waste rate decommissioning activities as low hazard (Rasmussen 1987).

6. 241-CX-72 TANK: SUMMARY OF HISTORY, CONTENTS, AND DECOMMISSIONING

REFERENCE DRAWINGS: H-2-2554 (BLUEPRINT)
SK-2-56955 (PIPING/MAP)

The Strontium Semiworks Facility was a pilot plant that tested proposed separation processes. Operated by Hanford Laboratories in the 1950s, it was used to evaluate full scale operations such as the reduction-oxidation (REDOX) and the plutonium/uranium extraction (PUREX) plants. The plant was also used for strontium recovery and americium and curium isotope recovery.

Tank 241-CX-72 was used as an experimental tank to study the characteristics of self-concentrating waste from the PUREX process at the Semiworks Facility. Made of 3/8-in. carbon steel (ASTM A-7-52T), the tank measures 40 in. in diameter and 35.67 ft in length (Figure 11). It rests on a concrete pad which forms the base of the enclosing caisson. The walls of the structure are enforced with five stiffener rings that extend nearly to the caisson wall. A cylindrical heater is located just above each ring and was used in the evaporation process. Three rows of vertical guides connect the rings. A 3-in. dry well, a 3-in. vapor header, and a 2-in. fill pipe are located at the top of the tank. Tank 241-CX-72 also has two 8-in. risers and instrument dip tubes. A manually operated system of paddles mounted on concentric rods extends the length of the tank and was used to "feel" the sludge.

The caisson is made of 1/2 in. carbon steel, and is 6 ft in diameter and 36.5 ft long. The concrete base of this structure is a 12-in. thick pad that is supported by reinforcing bars welded to the inside of the caisson. A single plate seals the tank and extends beyond the caisson edge.

The tank is located at coordinates N42058 and W50072.5, approximately 147.6 ft southeast of the 201-C building site in the 200 East Area (Figures 9 and 12). The top of the tank is 14 ft below grade. Installed in 1955, records indicate that the tank operated for less than one year (Ludowise 1989).

The last sampling found was from June 1974 (Ludowise 1989). The results were as follows:

Sludge quantity	73.5 in.
liquid quantity	1 in.
Plutonium	1.13 E^{-8} g/gal
Uranium	2.43 E^{-3} g/gal
Cesium 137	none detected
Strontium 89,90	4.33 nCi/gal

The records for the next several years indicate uncertainty in the level measurements and seem untrustworthy. A November 1974 reading placed the total level at 75.5 in., while a 1976 inspection indicated no sludge present. Then, in a confused 1977 measurement, a solids level of 1 in. and 325 gal was reported, and the liquid level was at 74.5 in. and 5 gal (Ludowise 1989). While it is probable that the volume and liquid measurements were accidentally

Figure 11. Simplified Cross Sectional View of 241-CX-72.

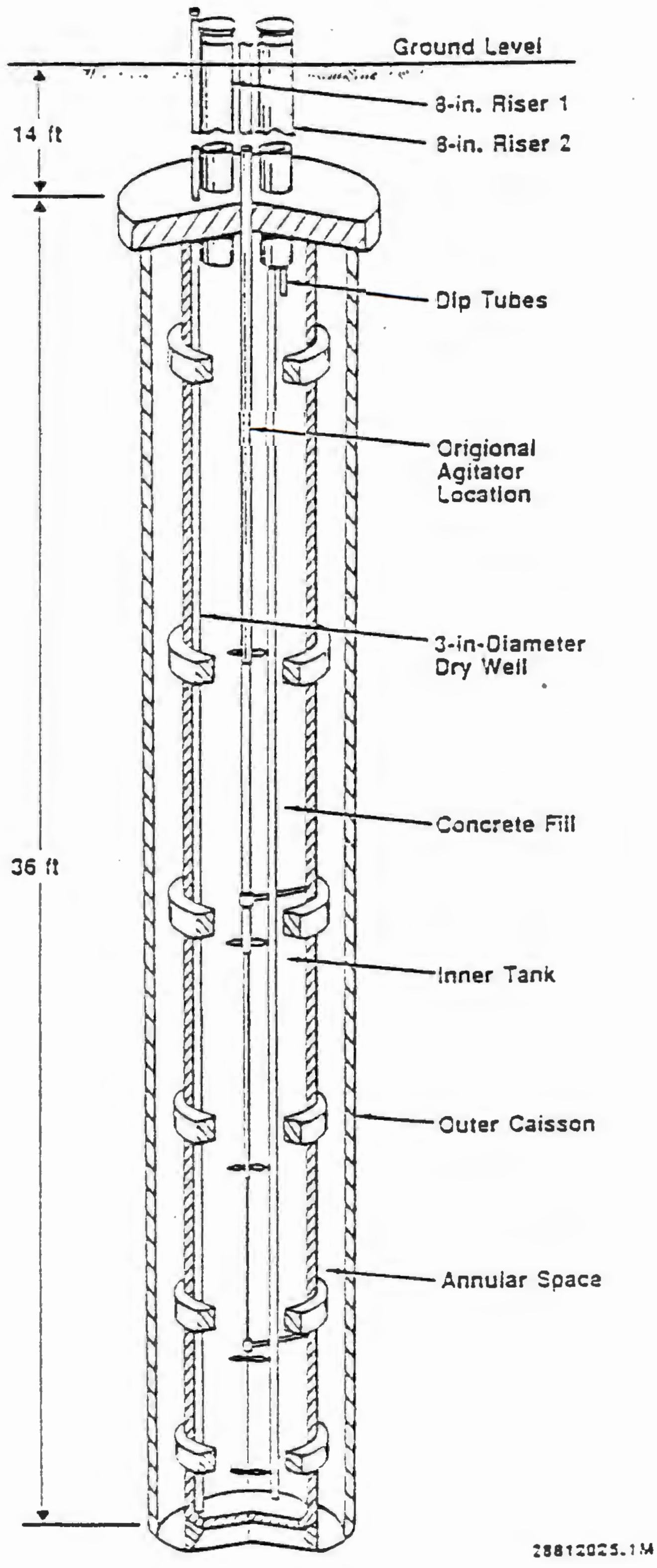

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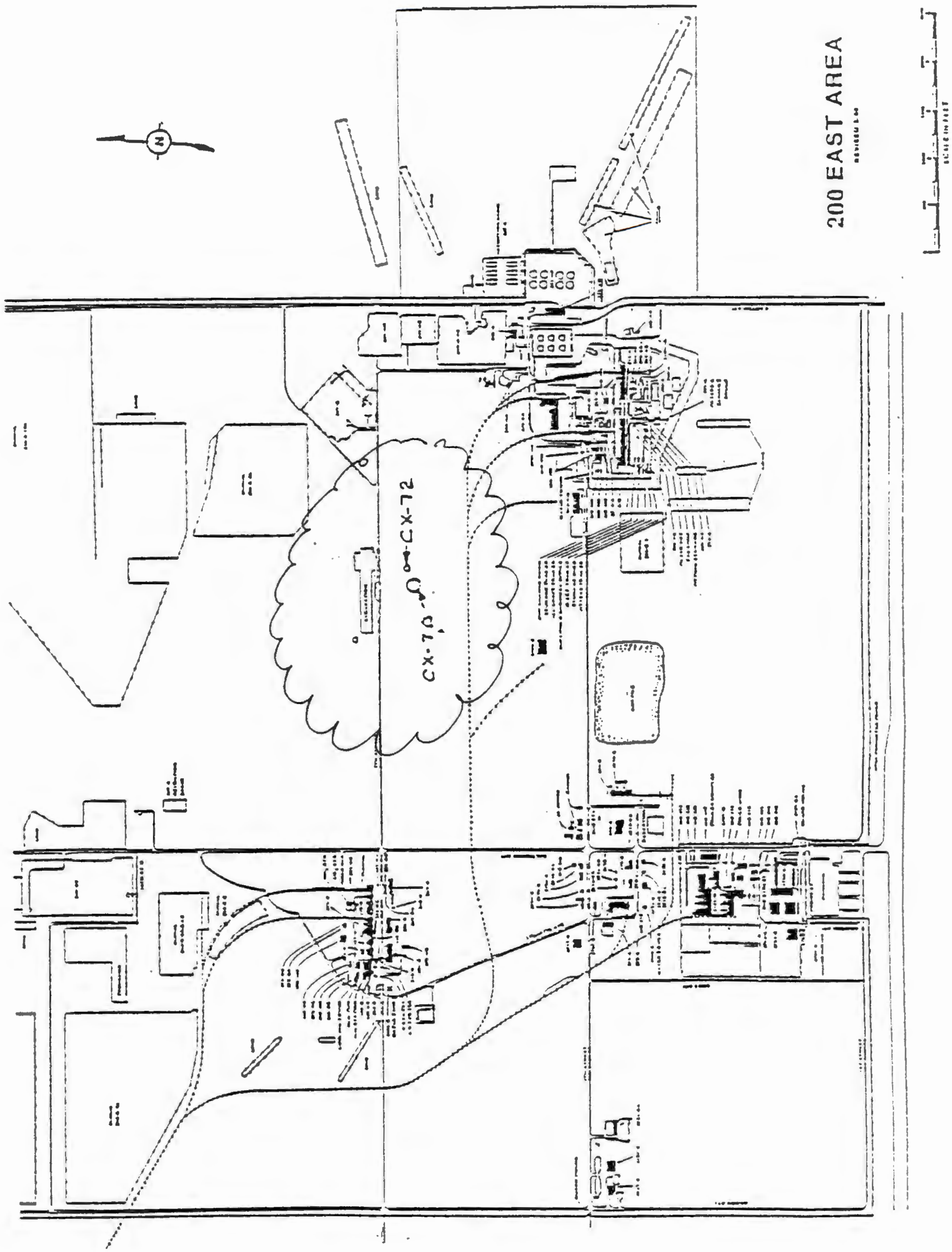
Figure 12. Plan of 200-E Area Showing the Location of Tanks at Strontium Semiworks.



reversed in the 1977 report, no information has been found to explain the discrepancies in any of these readings. By March 1978, the tank was recorded as being empty. A level measurement in 1986 supported this and the tank was filled with grout.

In October 1988, a rod from the paddle system was found to have been accidentally pulled approximately 15 ft out of the tank. The rod was determined to be contaminated and was removed and buried as low-level waste. Due to the suspicious levels of contamination on equipment from a supposedly empty tank, further study of tank 241-CX-72 was initiated.

Transuranic content was estimated using a neutron probe and radiation measurements through the 3-in. dry well. It was found that the tank had indeed not been empty when it was filled with grout.

It is believed that by 1986, internal heat generation had dried the waste and it had become very hard, similar to "salt cake." The final measurement that found the tank empty was probably in error for one of two reasons. Either the hard waste was mistaken for the tank bottom, or the reading was taken through the dry well.

Tests in 1988 found gamma radiation to be at 476 R/hr at a point 47 ft below grade (4 ft up from the bottom of the vessel). Thirty-eight feet below grade gave a measurement of 168 R/hr. Moving up to grade level, gamma radiation quickly dropped to 0.0006 R/hr (Ludowise 1989).

Neutron measurements were taken 12 ft from the bottom. It was found that predominately fast neutrons were present. The lack of thermal neutrons indicates that no moderator, such as water, was present in the area monitored (Ludowise 1990b).

The waste in the tank is thought to contain 150 to 200 g of plutonium-239. It is believed that most of the TRU radionuclides are present as fluoride compounds (WHC 1990). The tank is estimated to have a total of 6,000 Ci of beta/gamma (WHC 1987), although another document (WHC 1990) indicates that the cesium-137 could actually be as high as 10,000 Ci.

Assuming the tank received PUREX type wastes, the waste should not be hazardous, although this may not be the case if the tank also received flush chemicals.

The potential for a criticality has been evaluated (Ludowise 1989) and would not occur due to credible events. However, it is not recommended that the waste be exposed to water or other moderating materials until sampling has confirmed the lack of criticality potential (Ludowise 1990b). Neglecting nuclear safety concerns, tank 241-CX-72 should still not have water added because it is unknown whether or not the tank leaks.

Three alternatives for decommissioning were originally considered:

- In-place disposal
- In-place waste removal
- Tank removal and in-vitro waste removal

In-place disposal is prohibited by regulation and tank removal has a relatively high risk involved due to the need to move and section the tank. It was decided that in-place waste removal would be the best option (Cummings 1989b).

The current plan for decommissioning is broken down into three phases:

1) Grout removal
2) Sampling and analysis of sludge
3) Sludge removal as is found necessary

A more detailed breakdown can be located in WHC-SD-DD-PAP-002, "Task Plan for Sampling and Decommissioning of Tank 241-CX-72" (Ludowise 1990b).

As of July 1991, the program is in the middle of phase one. The site has been excavated to the top of the tank and covered with a greenhouse. A 36 in. diamond drill has been acquired to grind the grout so it can be vacuumed out (Ludowise 1990a).

The drilling apparatus is scheduled to be tested on a model section of the tank before the actual drilling begins at tank 241-CX-72. The testing is targeted to start in November 1991.

The grout is 100 to 350 mesh sand with 3/8 in. aggregate. Grout samples were tested with a penetrometer and had a calculated compressive strength of 520 to 945 psi. With the exception of the bottom 2 ft, the grout itself is believed to have little or no contamination.

A letter from Decommissioning Engineering (Speer 1991b) claims that the tank will be drilled to a depth of 22 ft, as estimates place the waste level at a depth of no less than 25 ft. To further assure that the contaminated grout is not drilled through, the grout will be monitored as it is removed. The drilling process will be stopped if contamination levels indicate that drilling is nearing or into the waste.

Once the grout is removed as far as possible, the waste will be core sampled. The results will be used to determine the method of the waste removal operations.

Total costs for decommissioning tank 241-CX-72 ranges from $4 million to $8 million, depending on the complexity of sludge retrieval. The entire project, which began in the fall of 1990, is expected to last approximately 4 years.

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7. 241-Z-8 TANK: SUMMARY OF HISTORY AND CONTENTS

REFERENCE DRAWINGS: H-2-16653

The 241-Z-8 tank is a horizontal cylindrical vessel located 160 ft due east of the 234-5 Building (Figure 13) and 6 ft below grade. Built in the 1950s, the tank took in waste from the 234-5 Building until April 1962.

The tank is 8 ft in diameter and 40 ft long (Figure 14). Resting on its side, 241-Z-8 has two fill pipes entering from the west end and one overflow pipe exiting from the east. All three pipes are 6 in. below the top of the tank, and the fill pipes have been blanked. Four additional pipes are located on the top of the tank. Two are 4-in. risers that extend above grade. The other two are larger openings which have bolted covers within 6 in. of the tank top (Irish 1974). With the exception of the fill pipes, it is unknown whether or not the pipes remain in their original, operable condition. Drawings indicate that the tank is surrounded by sand and is constructed of either 5/16 in. steel or wrought iron plates.

Tank 241-Z-8 was used as a solids settling tank for backflushes of the feed filters in the Recuplex process. Silica gel was used as a settling agent on the dissolved solids. The solids and the silica gel were then flushed to tank 241-Z-8 with nitric acid. The tank's overflow was sent to the 241-Z-8 french drain, which is approximately 36 ft to the east (Irish 1974).

In July 1959, records indicated that the tank was at its overflow capacity, which is 15,435 gal (Turner 1972). Between 1959 and 1962, no records were found to indicate that the tank had been pumped. Therefore it is possible that the tank was still at the overflow point when it was deactivated in 1962 (Irish 1974). Surveillance data from April 6, 1974 indicates a liquid content of 7,650 gal and a sludge content of 500 gal, for a total waste volume of 8,150 gal (Peabody, et al. 1974). This means that up to 7,185 gal may have been lost between 1962 and 1974. This possible loss has not been accounted for and no records found suggest a reason for the difference in the overflow value and the 1974 measurements.

Samples were taken from the 4-inch risers in April and May of 1974 and analyzed for plutonium content. The results of these analysis are shown in Table 6 and were used to calculate the maximum potential inventory at that time.

Tank pumping was initiated after preparation of a criticality prevention specification. Seven thousand six hundred gallons of waste and 5,000 gallons of flush solution were pumped from the tank and trucked to the 241-TX-109 tank. This left 7 inches of sludge (approximately 500 gallons) of waste in the tank. A sample of sludge taken beneath the 4-inch riser on October 22, 1974, contained 0.02 g/l of plutonium. This concentration calculates to a residual inventory of 38 grams of plutonium. The pH of the waste was 6.1 (Irish 1974).

Hanford Restoration Operations has no immediate plans for further decommissioning of tank 241-Z-8.

Figure 13. Site Facilities Map for Tank 241-Z-8.

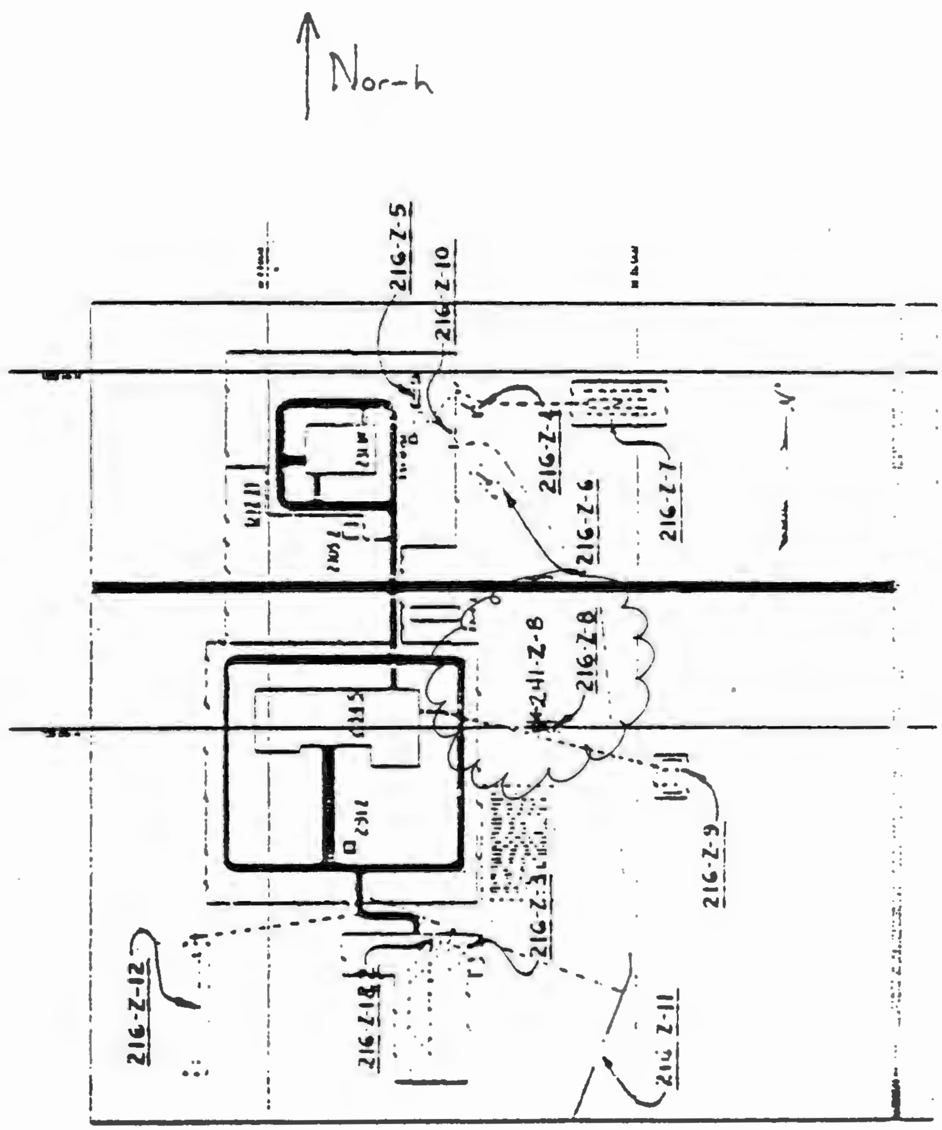

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Figure 14. Layout of Tank 241-Z-8.

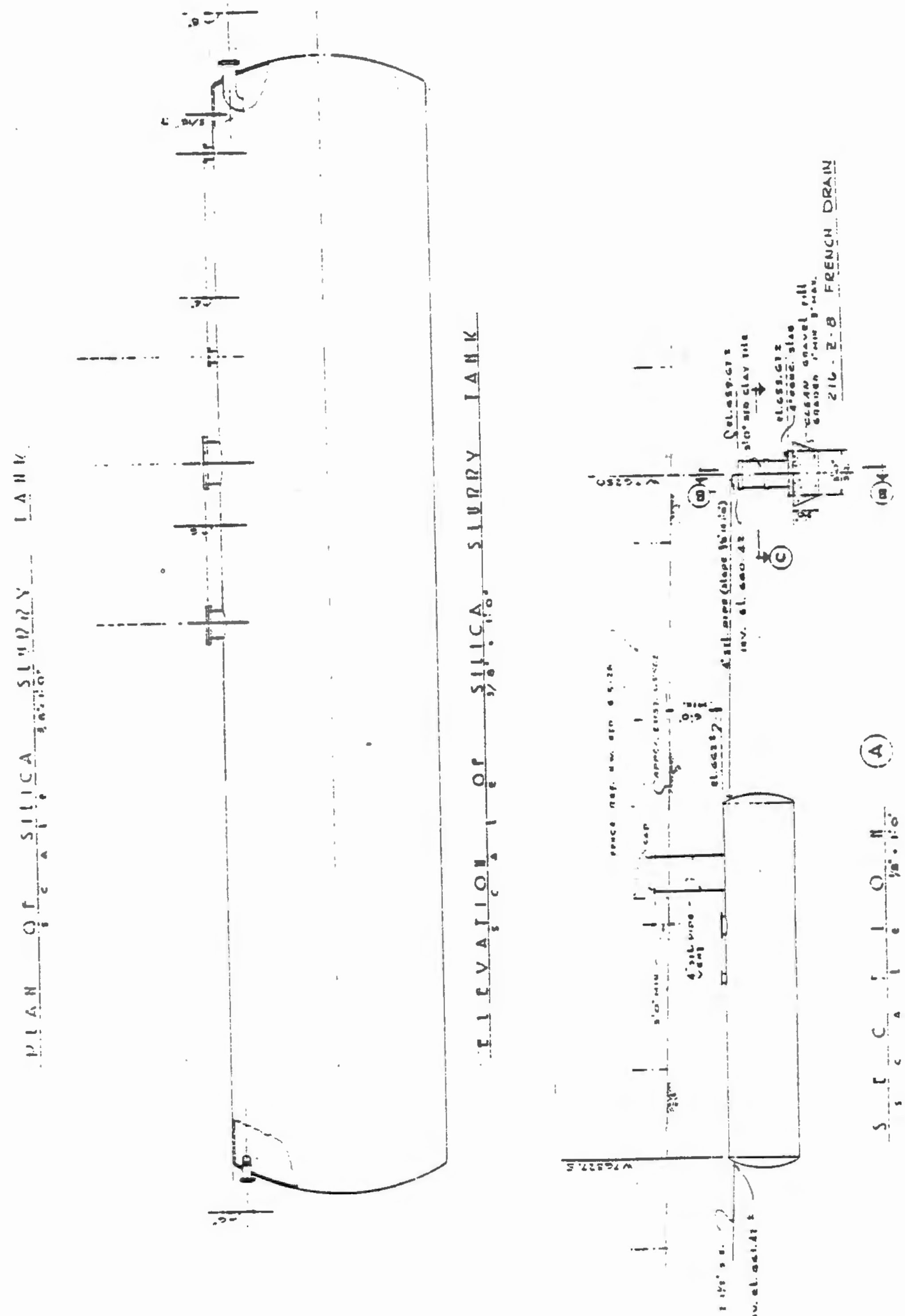

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Table 6. Plutonium Inventory of Tank 241-Z-8 Prior to Pumping.

	VOLUME	PLUTONIUM (grams/liter)	PLUTONIUM (grams)
Liquid	7600 gal	0.001	29
		0.004	116
		0.006*	174*
Sludge	500 gal	0.004	8
		0.25	475
		0.76*	1444*

* Maximum plutonium ~1600 grams
Liquid pH = 6
Liquid SpG = 1.034

8. 270-E-1 NEUTRALIZATION TANK: SUMMARY OF HISTORY AND CONTENTS

REFERENCE DRAWINGS: H-2-43110 (BLUEPRINT)
H-2-43118 (INSTALLATION PLANS)
H-2-44510 SHEET 97 (AREA MAP)

The 270-E-1 condensate neutralization tank is a stainless steel structure located at coordinates N42600 and W54400, approximately 300 yd west of the 221-B Building (Figure 15). It is 10 ft below grade. Reports indicate that its last period of service was from 1952 to 1957 (Wodrich 1991).

The tank is 9 ft in diameter and 9 ft high, with a sloping bottom and a capacity of 4,185 gal (Figure 16). A 41.5 in. charging tube in the center of the tank and a 6 in. sampling riser extend to 12 in. below grade. Both have been capped, along with the tank's inlet and outlet.

Tank 270-E-1 was originally constructed in the 1940s, and was used in conjunction with the 221-U Building. The tank was later removed, reworked, and put to use in 1952 to handle process condensate from B-Plant. Waste was pumped into the tank through the inlet at the bottom of the tank, traveled upward through a bed of limestone, exited through the outlet at the top of the tank, and then discharged to underground cribs.

Information on the tank is very limited and no sampling data has been found. Radiation readings have shown less than 100 counts/min direct and smearable and less than 0.5 mrad/hr penetrating plus nonpenetrating at the risers. The tank is thought to contain 3,800 gal of material (WHC 1987).

A report (Harlow 1974) indicates that the surface of the limestone was at 7.58 ft, and is described as having a gravel like texture. No liquid was visible. As the bottom of the overflow pipe is at 7.98 ft, at least 0.4 ft of liquid had evaporated since the tank was taken out of service. As all the entrances and exits to the tank have since been blanked, evaporation should no longer be a factor.

The liquid level remains unknown. Considering that limestone dissolves only slightly in water, the contents of the tank should have remained relatively unchanged since 1974. With the limestone level at 7.58 ft and assuming a 40 percent void space, the tank could contain as much as 1,400 gal of liquid (Harlow 1974).

It is unknown whether or not the tank leaks, although there is no evidence that the tank has deteriorated.

No surveillance data has been compiled on tank 270-E-1 and HRO has no immediate plans for decommissioning.

Figure 15. Plan of 200-E Area Showing Location of Tank 270-E-1.

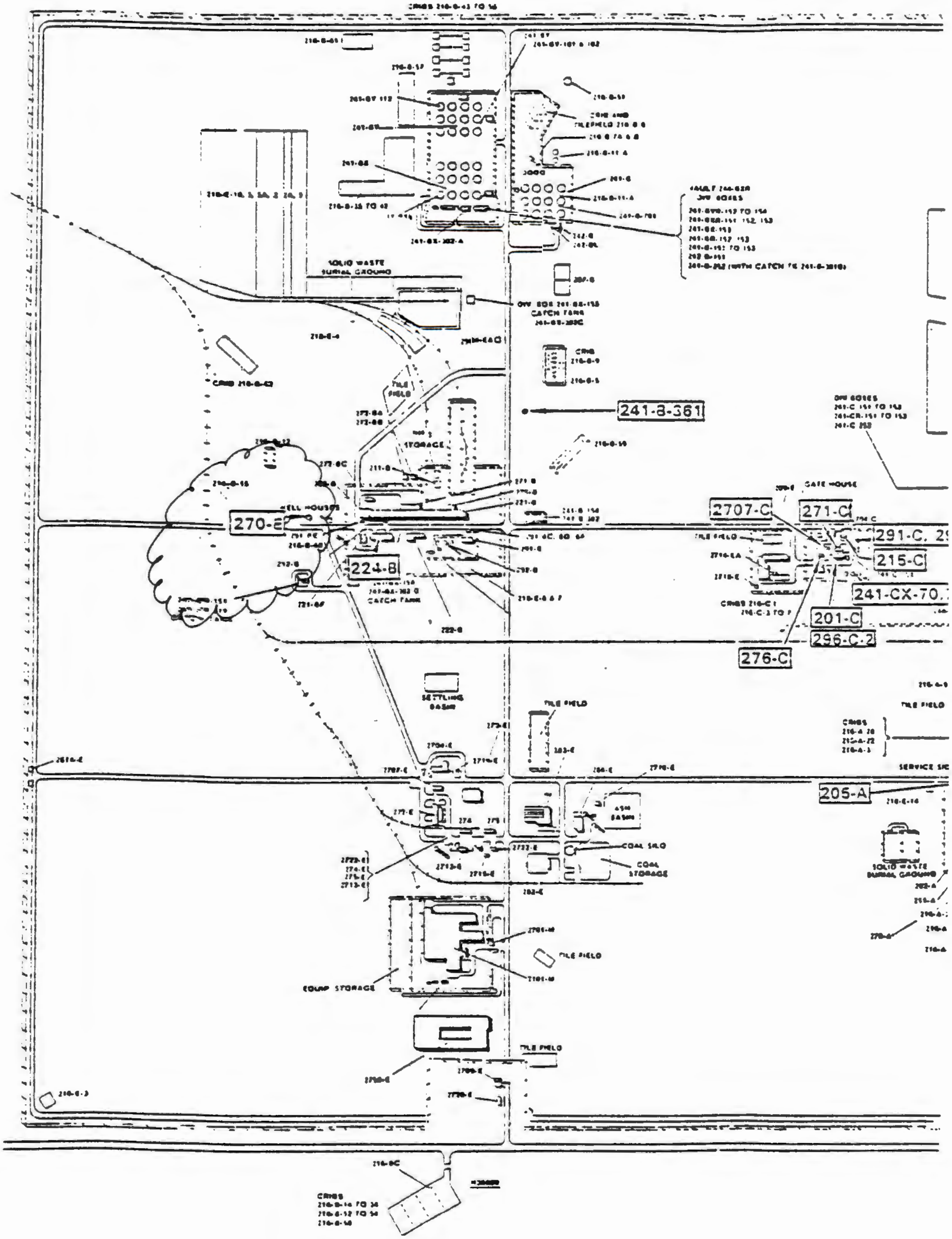

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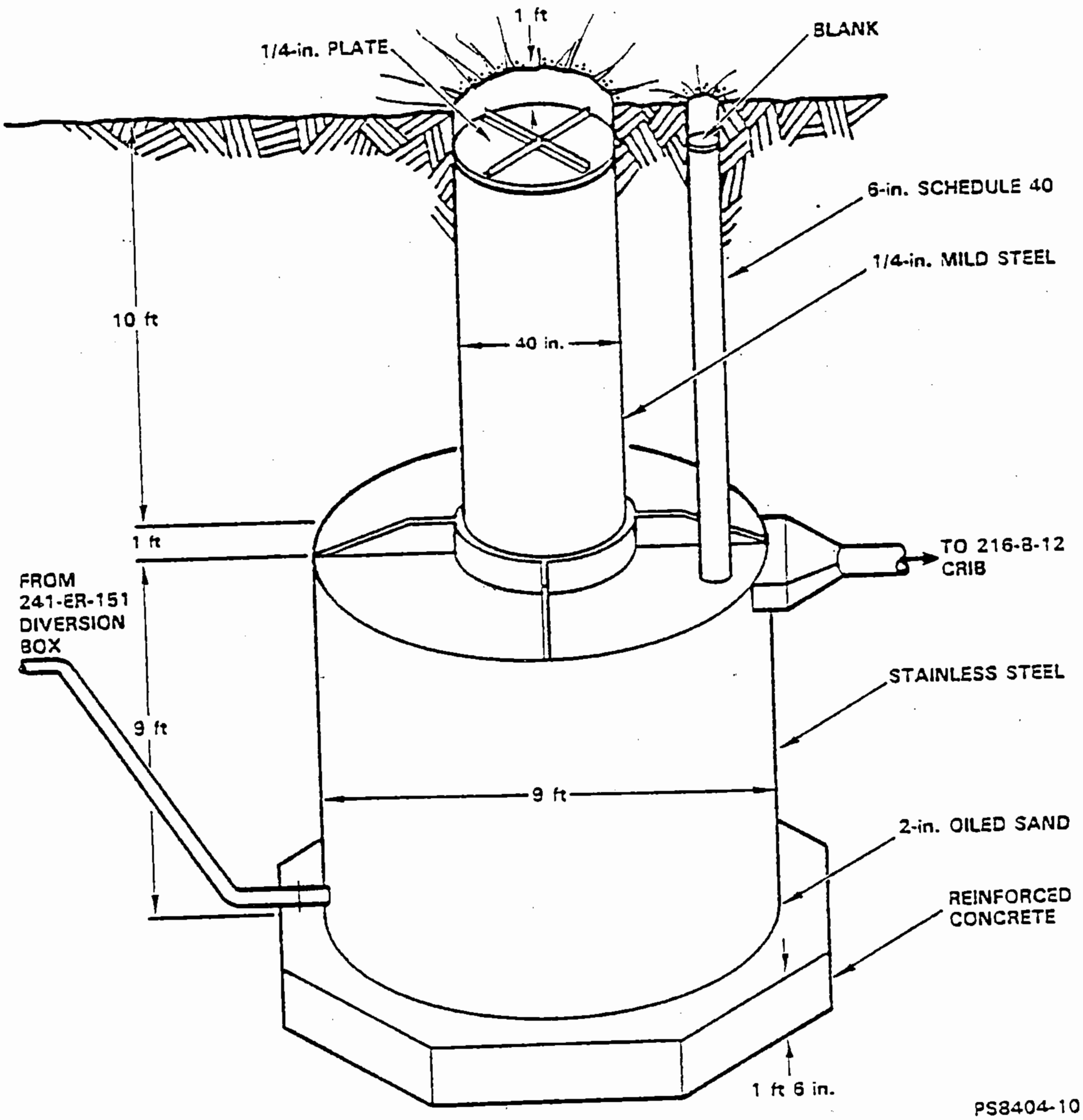
Figure 16. 270-E Condensate Neutralization Tank.



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9. 205-S PROCESS VAULT: SUMMARY OF HISTORY, CONSTRUCTION, AND DECOMMISSIONING

REFERENCE DRAWINGS: H-2-39655 (BLUEPRINT)
H-2-39654 (PIPING/MAP)
H-2-5140 (AREA MAP)

The 205-S process vault was used in conjunction with a group of structures that include the 205-S building, the 203-S Uranyl Nitrate Hexahydrate (UNH) storage tank farm, and the 204-S Tank Farm and Pumphouse (Figure 17). These facilities were constructed in the early 1950s and were used from 1952 to 1967 to decontaminate UNH from REDOX (UNH from PUREX was later brought to the site by truck).

The 205-S vault was the primary process unit for the decontamination process at the site. The concrete structure was 12 ft by 17 ft by 18 ft deep, and was located approximately at coordinates N34900 and W74270 in the 200 West Area. Inside the cell were absorption column SG-1 and neutralization tank SG-2.

The absorption column was 3.5 ft in diameter and 15 ft long. It was constructed of 1/4 in. stainless steel, had 1 in. inlet and outlet pipes, and was contained in a 5.5 ft (outer diameter) removal cask. The SG-1 column was filled with silica gel and was use to process ruthenium out of the uranium. Periodically, the uranium was removed from the column by flushing the column with 2,000 gal of water and 2,000 gal of nitric acid. The column would then be regenerated with 6,000 gal of hot dilute oxalic acid and flushed with 4,000 gal of water (Harlow 1974).

The neutralization tank was 10 ft in diameter by 10 ft high, with a capacity of 5,000 gal. Also constructed of 1/4 in. stainless steel, SG-2 tank was equipped with cooling coils, an agitator, and jets for pumping liquid out. After the fission products were collected and stripped by the column, they would be neutralized in the tank before being sent to underground cribs.

Upon closure of the REDOX plant, 205-S was placed on long-term standby. The silica gel absorption column was flushed with 14,000 gal of water, and the vault was left unused for over a decade.

In 1983, decontamination and decommissioning (D&D) efforts on the entire facility were begun by Rockwell Hanford Operations (RHO). The area was chosen for decommissioning due to its high surveillance and maintenance costs, moderate projected exposure levels to D&D workers, good cost to environmental benefit ratio, and the workable size of the plant.

The vault itself was found to have accumulated a substantial amount of liquid, which was presumed to be rainwater that had leaked through the cracks in the cell's cover blocks (Rasmussen 1984). This assumption is supported by a report (Harlow 1974) which states that in 1974 the tank had already accumulated 1,650 gal of liquid that was probably rainwater. This liquid was removed by RHO using a submersible electric pump and trucked to 241-SY-102 for evaporation. The liquid remaining in the column and tank were also pumped out shortly after.

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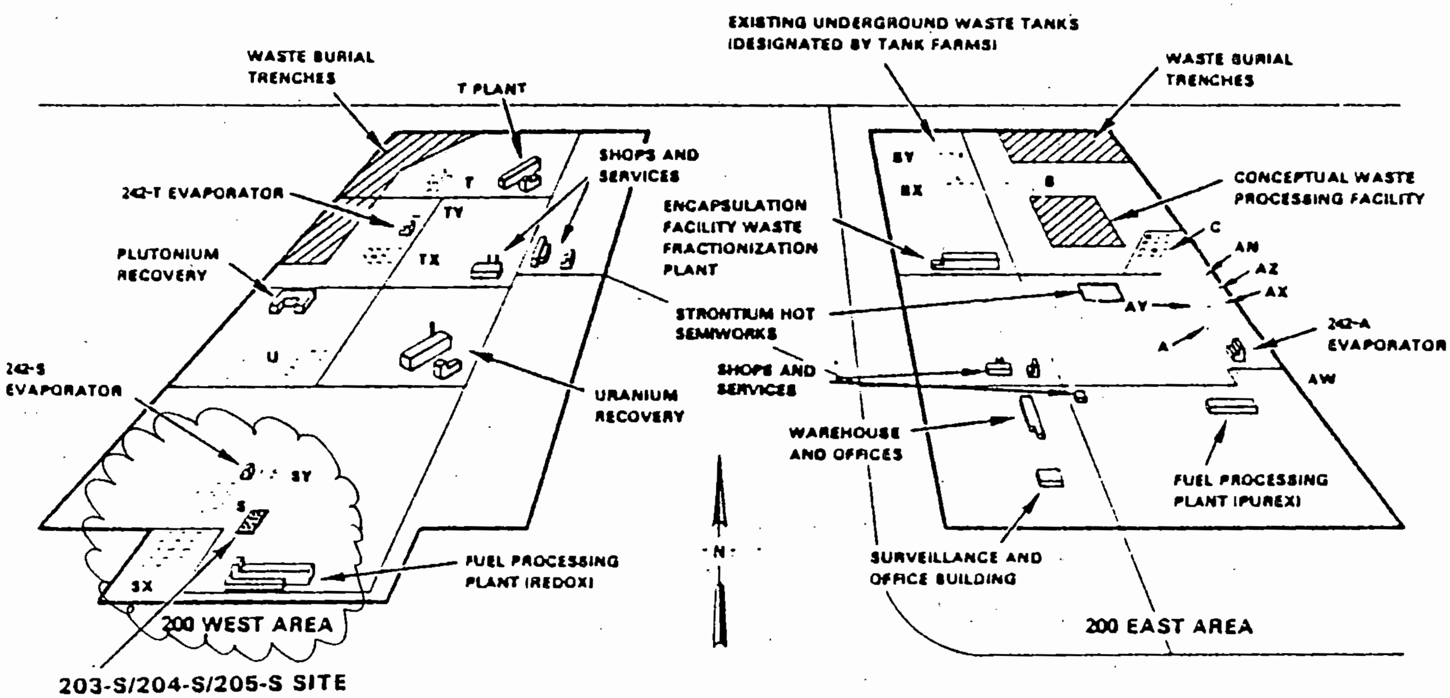

Figure 17. The 203-S, 204-S, 205-S Facility Location.

The neutralization tank was stripped, loaded on a truck, and stored in a regulated spare equipment yard for later use.

The SG-1 column also needed to be stripped before being disposed of. It was lifted to grade level, and the contents were allowed to drop to the cell floor. It was then wrapped in plastic and buried in the radioactive waste burial ground with all the other solid waste from the project.

The vault's cover blocks were placed in the cell, and backfill was poured to the level of the wall pipe nozzles. A 12-in. thick concrete pad was poured to seal all of the nozzles, and the cell was filled to the top with backfill. Another concrete pad was then poured over the area to keep any more rainwater from entering.

After the entire facility was decommissioned, any contaminated soil that could not be covered adequately was removed and transferred to the burial ground. The entire area was then covered with 2 to 10 ft of clean soil. Concrete markers identifying underground contamination were put into place and the area was seeded with wheatgrass. An estimated 0.2 Ci of fission products remains buried at the site. No further work is contemplated for this site.

The decommissioning project cost $1.025 million and lasted 10 months, from March to December 1983. It was completed both under budget and ahead of schedule.

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