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A Decommissioning Plan for the Heavy Water Components Test Reactor

by

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A DECOMMISSIONING PLAN FOR THE HEAVY WATER COMPONENTS TEST REACTOR

by

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CONTENTS

FOREWORD

Preparation of a decommissioning plan for the Heavy Water Components Test Reactor (HWCTR) was authorized by ERDA-DWMT as part of the FY 1975 budget. The results of the planning effort are documented herein.

A DECOMMISSIONING PLAN FOR THE HEAVY WATER COMPONENTS TEST REACTOR

SUMMARY

Three alternatives to decommission the Heavy Water Components Test Reactor (HWCTR) have been analyzed as summarized in Table 1. The protective confinement approach is advantageous as long as current activities onsite limit access by the general public; excellent confinement of the residual activity is provided by *in situ* dry storage as the radiation from ^{60}60 diminishes. Entombment provides the most-secure confinement of the activity but at some increased cost. Dismantling HWCTR has no apparent advantages other than a demonstration at the Savannah River Plant site, because of the long-term commitment to safeguarding radioactive material; the relative cost is high.

The induced radioactivity in HWCTR is currently 2.3×10^4 Ci; general area radiation levels are typically 3 mR/hr. In 35 years, the decay of ^{60}Co will lower the radiation levels by a factor of 100, and the remaining radioactivity will be 2×10^3 Ci of ^{63}Ni. Minimal offsite effects are calculated to result after postulated structural failures to the decommissioned HWCTR facility.

Flexibility and aesthetics favor dismantlement, but these criteria are considered less significant than public radiation dose, cost, and land area committed.

TABLE 1

Summary of Decommissioning Options

	Dismantlement	*Entombment*	*Protective Confinement*
	Remove all radioactive equipment to burial ground	Relocate above-grade equipment to burial ground, and fill building with concrete	Repair dome, seal building, and seal piping system
Radiation Exposure			
Accidental to Public	Unlikely	Extremely unlikely	Very unlikely
Planned Occupational, rem	20	5	<1
Land Area, acres			
HWCTR	0	1.5	1.5
Burial Ground	2	0.2	0
Water Rights, acres	Burial site to creeks (already needed)	From 0 to 90 (HWCTR to creek)	90 (HWCTR to creek)
Capital Cost, $ millions[a]	5.4	1.6	0.19
Annual Cost, $	<100	1500	3000
Flexibility	Best	Poor	Good
Aesthetics	Best	Good	Least attractive

a. Current appraisal of cost for evaluation only, based on January 1978 authorization.

GENERAL DESCRIPTION AND OPERATING HISTORY

The Heavy Water Components Test Reactor (HWCTR) was operated from October 1961 to December 1964 to test fuel elements and other reactor components of potential use in heavy water moderated and cooled power reactors. Operations were terminated, and the facility was placed in standby condition as a result of the decision by the U. S. Atomic Energy Commission to redirect the research and development work on heavy water power reactors to reactors cooled with organic materials. For about one year, the facility was maintained so that operation could readily be resumed. Subsequently, the facility was retired in place with monthly surveillance by reactor personnel from a nearby facility.

The HWCTR site is located in U area, which is three miles from the nearest major production site at the Savannah River Plant (SRP), and also about three miles from the nearest plant boundary. The area outside the plant boundary within a radius of ten miles of U area is sparsely populated, particularly within two or three miles of the Savannah River.

The location of the HWCTR facilities within U area is shown in Figure 1. An aerial photograph of the site is shown in Figure 2.

A cutaway view of the containment building is shown in Figure 3. The building is 70 ft in diameter and 125 ft high. Approximately half of the building is below grade and is prestressed concrete; the upper half is carbon steel. The building is designed to withstand an internal pressure of 24 psig and was tested pneumatically at 29 psig. The containment building houses the reactor and coolant systems, the charge-discharge mechanisms, and the reactor instrumentation.

A cutaway view of the reactor pressure vessel is shown in Figure 4. The vessel has an overall height of 30 ft, with a core-region height of 10 ft and a diameter of 7 ft. The shell and head are carbon steel; all interior surfaces are clad with stainless steel, 0.25-inch nominal thickness. The 3-inch radial thermal shield is stainless steel.

The core consists of a central region of 12 test assemblies surrounded by a ring of 24 driver fuel assemblies, enriched in ^{235}U. Control rods, safety rods, and instrument thimbles were interspersed throughout the core.

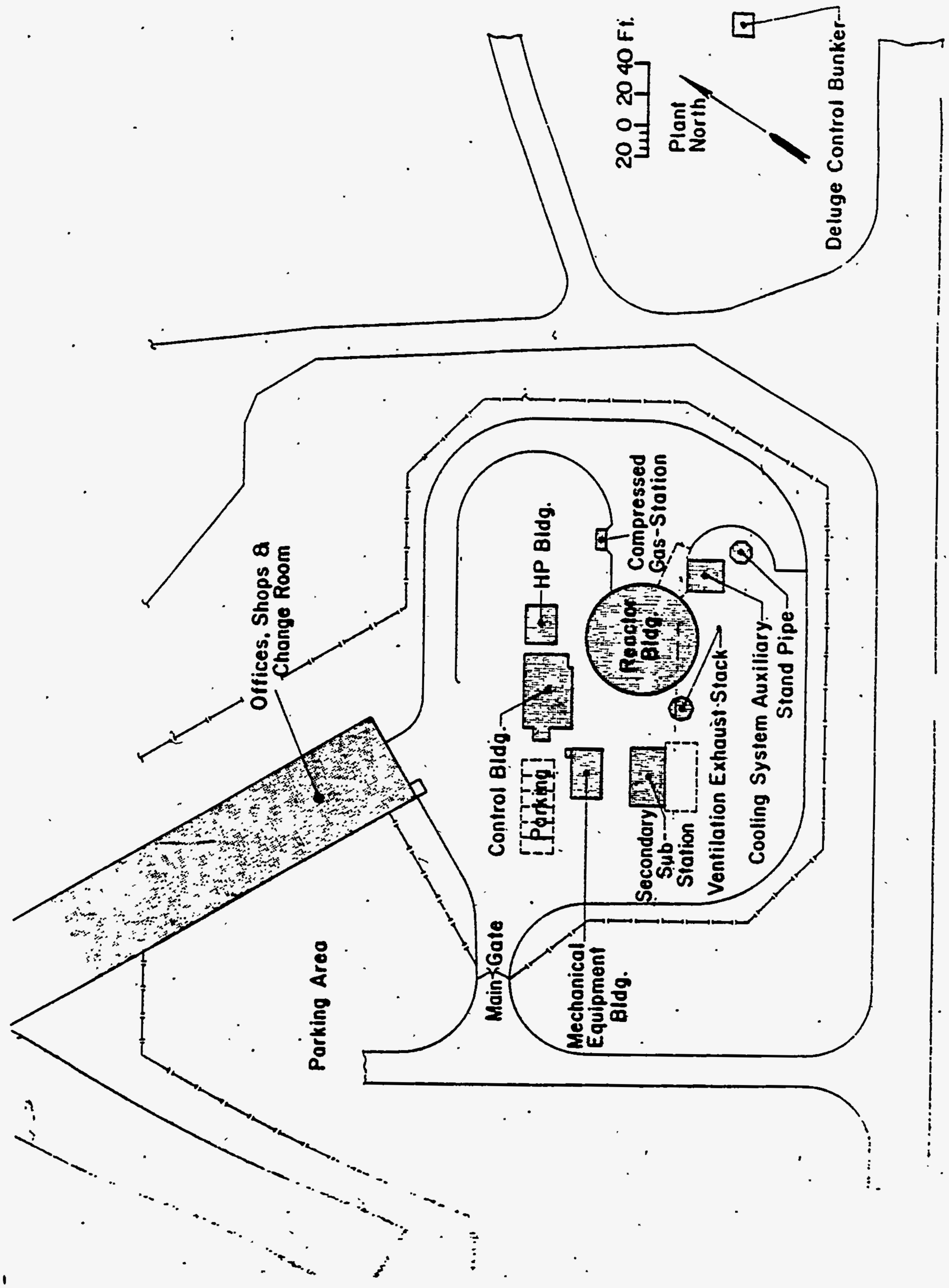

FIGURE 1 U-Area Building Arrangement

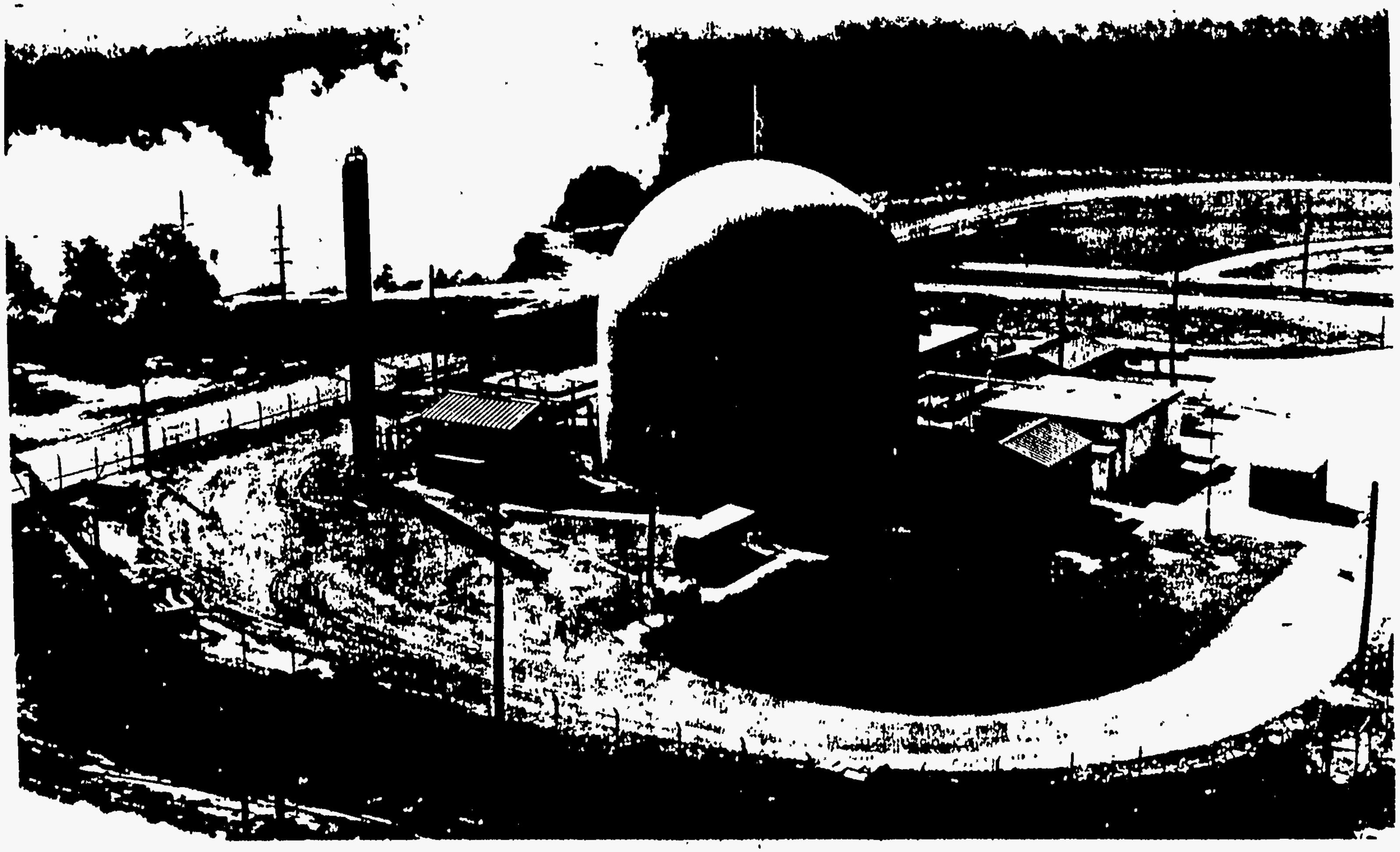

FIGURE 2   HWCTR Containment Dome and Auxiliary Buildings

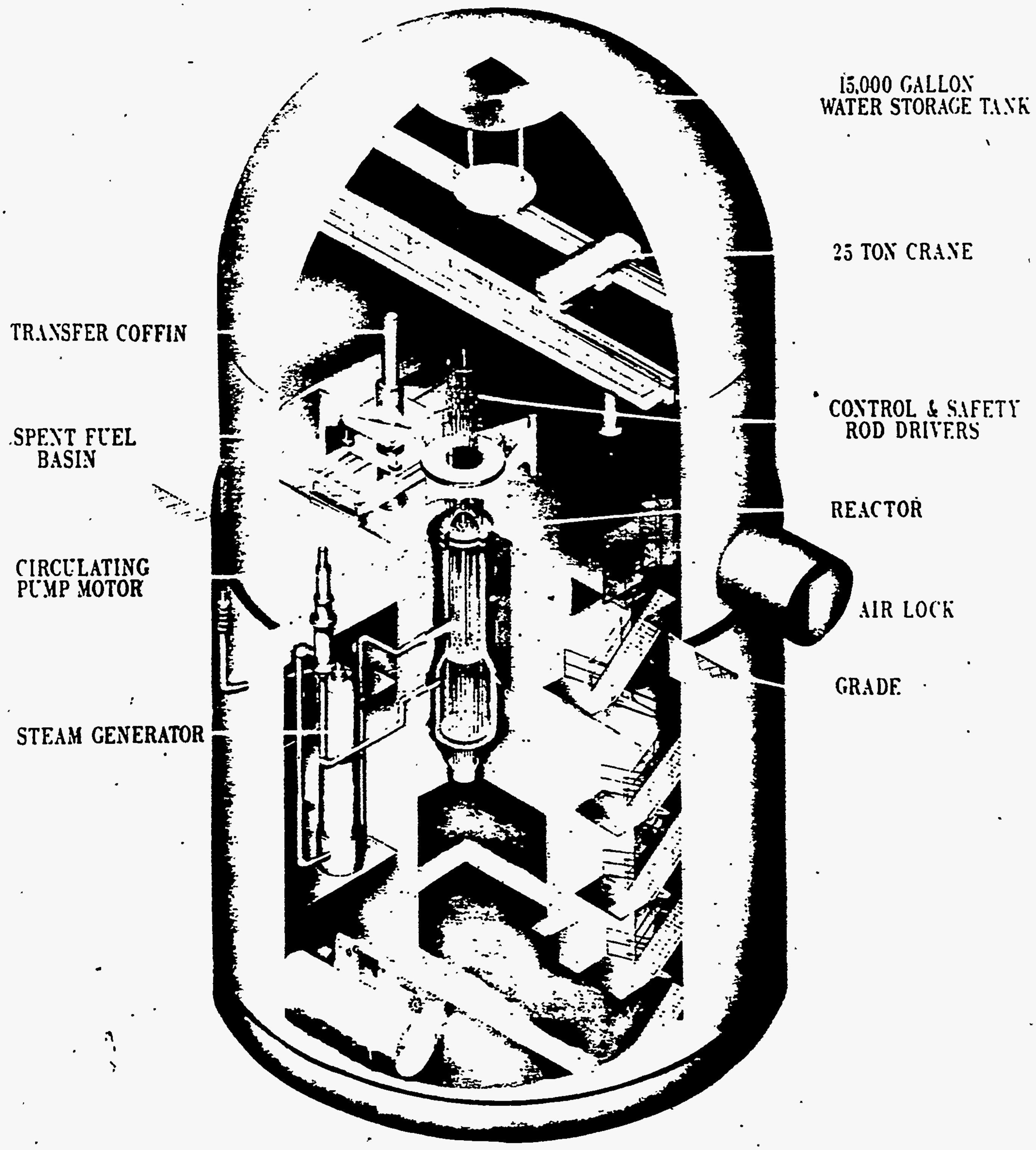

FIGURE 3. Containment Building

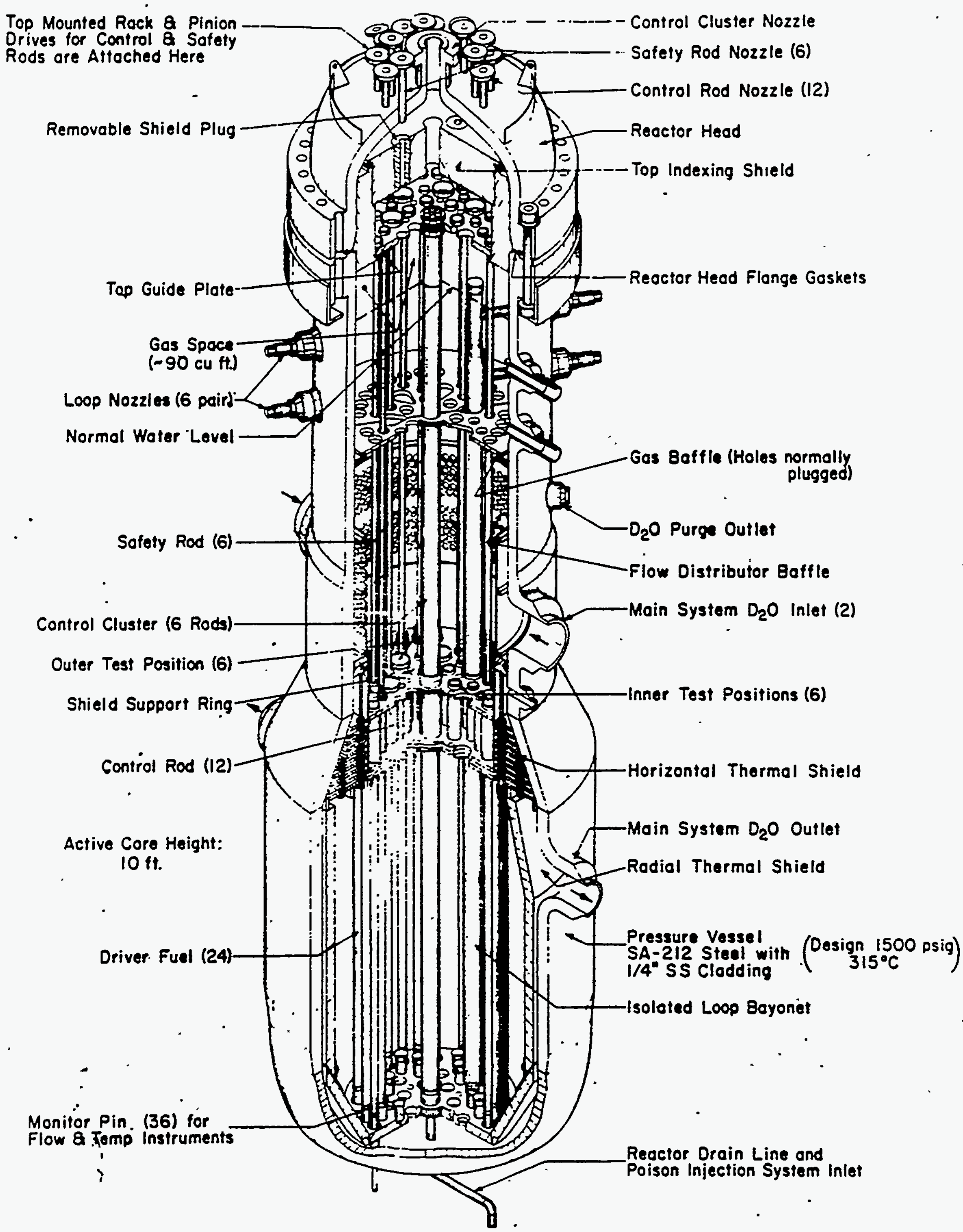

FIGURE 4. Reactor Vessel

Heavy water to moderate fast neutrons and to cool the fuel was pumped through two inlet nozzles into the top section of the reactor vessel at a rate of 10,000 gpm. The flow path is down through the fuel assemblies, up through the moderator space, and out into two coolant loops. The operating conditions in the heavy water system were 1200 psig at 230 to 250°C; the reactor vessel was tested at 1500 psig and 315°C. The heavy water was cooled by boiling light water in a steam generator in each loop; the steam produced was discharged to the atmosphere.

Two test positions in the reactor core were occupied by bayonets connected to isolated coolant loops called the "Liquid Loop" and the "Boiling Loop." A bayonet consisted of two concentric tubes that formed annuli for the downward (inlet) flow and upward (outlet) flow of coolant through the test assembly. The upper section of the bayonets and part of the external piping are made of stainless steel. The in-core sections are made of Zircaloy. The boiling loop bayonet failed when vibration caused a dummy housing to fret a hole in the bottom of the bayonet. The boiling loop bayonet never contained test fuel; the liquid loop bayonet contained a test assembly that failed and released particulate material.

During the initial hydraulic tests with light water, a protective film of magnetite was formed on the surface of the process system. This film remained intact during the subsequent operation and, together with careful alkaline pD control, accounted for the completely satisfactory performance of the large amount of carbon steel in the process system. Crud levels in the process system were very low, resulting in correspondingly low transport of activated particles.

The total nuclear exposure in HWCTR was 13,882 MWD. Maximum reactor power was 50 MW. Thirty-six test assemblies containing tubular fuel of uranium metal or uranium oxide were irradiated, and the utility of this fuel for power reactors was successfully demonstrated. One assembly of tubular oxide elements reached an exposure of 17,500 MWD/Tonne. Ten failures of experimental fuel were experienced during this period. In each instance the failure was detected promptly, and the reactor was shut down before the process system became seriously contaminated.

All failed fuel assemblies were transferred to the shipping cask under water in the spent fuel basin. As a result, some contamination of the basin occurred that was not completely removed by filters and ion exchange columns.

After the facility was shut down in 1964, all of the fuel assemblies and the two neutron sources were removed from the reactor and stored in H Area. All other reactor components including control and safety rods, long-term corrosion coupons, and a rod containing gamma ion chambers were left in the core. After the heavy water was drained from the reactor system, both the high and low pressure systems were dried and filled with nitrogen. A positive pressure was maintained from nitrogen-filled cylinders until this operation was abandoned in November 1971.

Very few changes have been made in the HWCTR system since shutdown. Radioactivity levels have decayed to low values with only a few isolated hot spots remaining. ^{60}Co is the primary detectable activity. The external appearance of components inside the containment dome is very good. The physical location and status of the HWCTR system are essentially the same as described in the standby status report.

HWCTR RADIOACTIVITY

The definition of the activity present at HWCTR was subdivided into three general tasks:

- Estimate the residual activity in the carbon steel portion of the reactor hydraulic system exterior to the biological shield.
- Estimate the induced activity in the reactor vessel and adjacent concrete.
- Survey the radiation emitted in various work areas in the facility.

The activity in the carbon steel piping is shown to be small (less than 0.01% of the total) and is from deposits of activated corrosion products and adsorbed activity from fuel failures; the HWCTR test failures are described. Calculations of the activity ($\sim 10^4$ Ci) induced in the reactor vessel (primarily the thermal shield) from flux and exposure data are corroborated by some measurements made near the vessel with special instruments (Appendix A). The good agreement between calculations and measurements supports the preliminary estimates of the induced activity in materials near the reactor. Survey data of HWCTR were used in estimating occupational doses in decommissioning. Radiation surveys of the area are compared in Appendix A, Table A-1.

System Activity Inventory

Fuel Failures

Ten test fuel assemblies (Table 2) failed during operation in HWCTR. Only two of the failures resulted in the release of uranium to the main coolant system; one failure occurred in the isolated liquid loop system.

The amount of uranium released in failures 1, 2, and 8 is uncertain. Values used in this study were 1, 5, and 5 g, respectively. The amounts of plutonium and fission products contained in a gram of uranium were calculated from the exposure history of the assembly. Based on these calculations, approximately 4 mg of plutonium (predominantly ^{239}Pu) was released to the main process system and 4 mg to the liquid loop during HWCTR operation. The corresponding values for fission products are 6 mg (main system) and 5 mg (liquid loop).

TABLE 2

Failed Test Fuel Assemblies

Assembly No.	Acronym[a]	*Type of Fuel*	*HTR Report[b] Number*	*Estimate of Released Uranium, g*
1.	TWNT-7	Natural U Metal	45	0.1 to 12
2.	TWNT-14	Natural U Metal	47	0.5 to 60
3.	SOT-2-3	Natural U Oxide	51	Gaseous
4.	SOT-2-2	Natural U Oxide	54	Gaseous
5.	OT-1-6	1.5% Enriched U Oxide	55	Gaseous
6.	OT-1-3	1.5% Enriched U Oxide	56	Gaseous
7.	SOT-5-2	Natural U Oxide	61	Gaseous
8.	SOT-7-2	Natural U Oxide (Liquid Loop)	65	About 5
9.	SOT-9-2	1.2% Enriched U Oxide	72	Gaseous
10.	CANDU	Natural U Oxide	76	Gaseous

a. TWNT - thin-walled nested tube
SOT - segmented oxide tube
OT - oxide tube
CANDU - fuel planned for AECL CANDU reactor (Canadian deuterium oxide cooled and moderated, uranium fueled reactor)

b. A separate HTR (HWCTR Technical Report) was written for each failure and is in the HWCTR permanent files.

The response of the fission-gas monitors during the SOT-7-2 failure in the liquid loop was much greater than for the two metal failures, although the total uranium released was about the same. The high gas activity from SOT-7-2 is attributed to molten uranium penetrating the fission-gas collection chamber in the failed segment and driving out accumulated gases.

Considerable difficulty was encountered in reducing liquid loop activities after the failed assembly was discharged. Deionizer failure and pluggage occurred, and hot spots were found in several parts of the system. Deionizer replacement and special flushings were only partly successful in lowering the activity levels.

Radiation surveys made after reactor shutdown revealed high activity regions in the boiling loop inlet and outlet stubs. Recent surveys confirm the presence of above-background activity. Although the nuclides have not been identified, it is very likely that the radiation is originating from activation products deposited in the bayonet and stubs during the period prior to detection of the bayonet failure. The bayonet contained an empty Zircaloy housing resting on a stainless steel cross. Vibration resulted in severe damage to the housing, cross, and bayonet itself, until fretting wore a hole in the bayonet. D_2O in the bayonet had been isolated from main process D_2O and was stagnant for a period of at least 4 months before the bayonet failed in August 1963. The D_2O was known to be very turbid and to contain corrosion products of Zircaloy and stainless steel. Some of this material was probably activated in the reactor core region and migrated to and deposited on the inlet and outlet stubs, which were closed by blind flanges. The stubs were not flushed after the failed bayonet was removed.

Piping Survey

Samples were obtained from both of the liquid loops and the main process water header at the lower side of the first accessible horizontal run after leaving the reactor vessel. Smears were also taken from the bottom of the process water storage tank, EP-41. Results are summarized in Table 3; details are given in Appendix A. Less than 0.01% of the total HWCTR activity is deposited in the pipes. All pipe samples represent high side estimates of the radioactivity deposited in the system. However, the radioactivity measured in the boiling loop is puzzling because no assemblies were tested in the loop. As discussed previously, a portion of the boiling loop was blanked after a bayonet failure. It is believed that activity from a fuel failure in the liquid loop subsequently contaminated the main system and the boiling loop via common piping.

TABLE 3

Radioactivity in the External Water Systems

	Radioactivity, Ci		
	TRU[a]	^{137}Cs	^{60}Co
Main Process Water System	170×10^{-6}	$<0.6 \times 10^{-6}$	0.5
Liquid Loop	230×10^{-6}	$<14 \times 10^{-6}$	0.05
Boiling Loop	380×10^{-6}	$<35 \times 10^{-6}$	0.32
EP-41	26×10^{-6}	11×10^{-6}	0.70×10^{-3}
	806×10^{-6}	11×10^{-6}	0.87

a. ~90% ^{239}Pu, 10% ^{238}Pu (suspected to be <1% ^{238}Pu)

Neutron Induced Activities

Calculations were made to determine the activity levels of several major system components exposed to neutron flux irradiation during reactor operation. Representative radial and axial distributions of the thermal neutron flux are shown in Figures 5 and 6. The axial flux distribution is strongly peaked near the bottom of the reactor because full length control rods were operated as a bank, and no partial length rods were used for flux shaping. Absolute values for the radial flux correspond to a total reactor power of 50 MW midway through a fuel cycle. The axial location corresponds to the layer of maximum axial flux. **The neutron flux in the concrete biological shield is reduced a factor of 10 for each 9 inches of concrete. The shield wall between the reactor and the stairwell is about 11 ft thick.**

Radioactive Nuclides

A review of potential activation products in stainless steel and carbon steel showed that only three nuclides contribute significantly to induced activity 10 years after reactor shutdown, viz., [illegible]Fe, [illegible]Co, and ^{63}Ni. Because the total fluence (ϕt) in regions of interest was relatively low, no multiple neutron captures need to be considered in the formation of any of the three nuclides. Rather, the concentrations can be calculated from nuclear data related only to the nuclide and its immediate precursor. Activities were calculated from thermal neutron fluxes and 2200 m/sec cross sections without a Maxwellian-distribution term; reactions from epithermal neutrons were not treated explicitly.

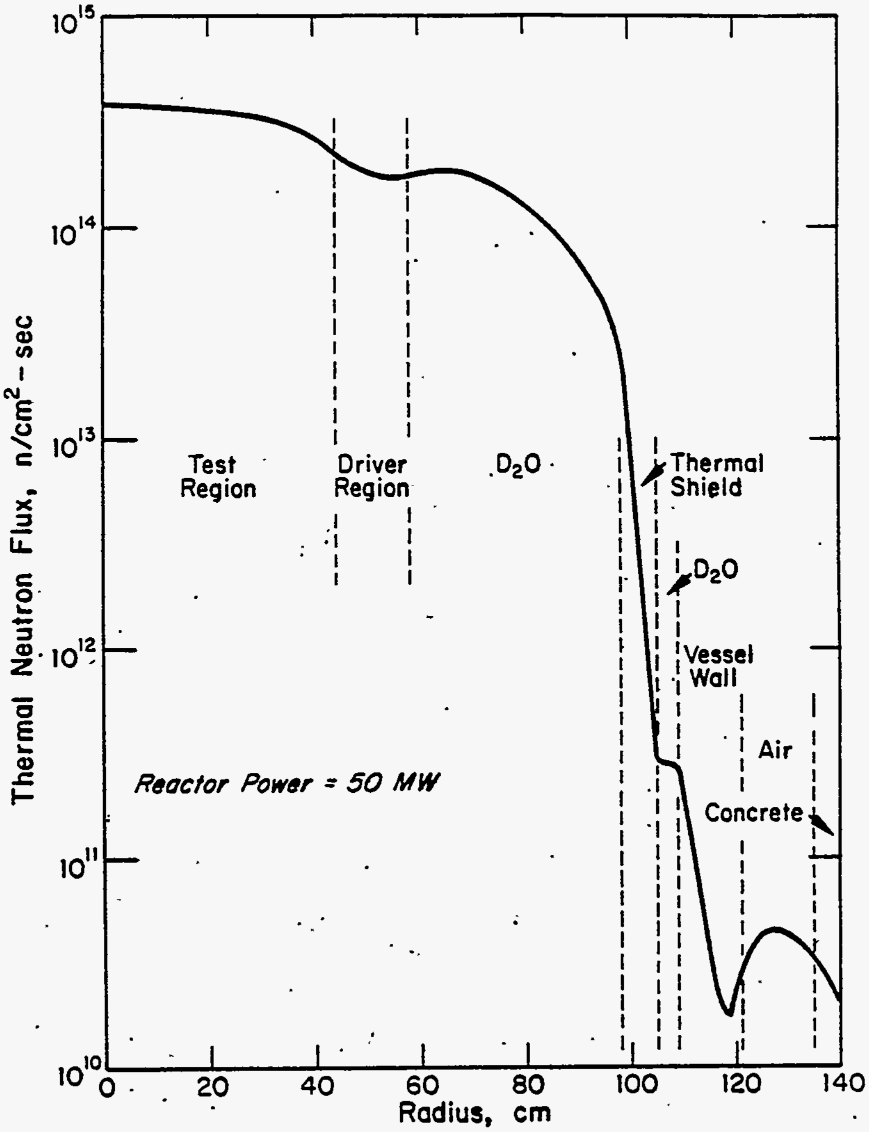

FIGURE 5. Radial Profile of Thermal Neutron Flux

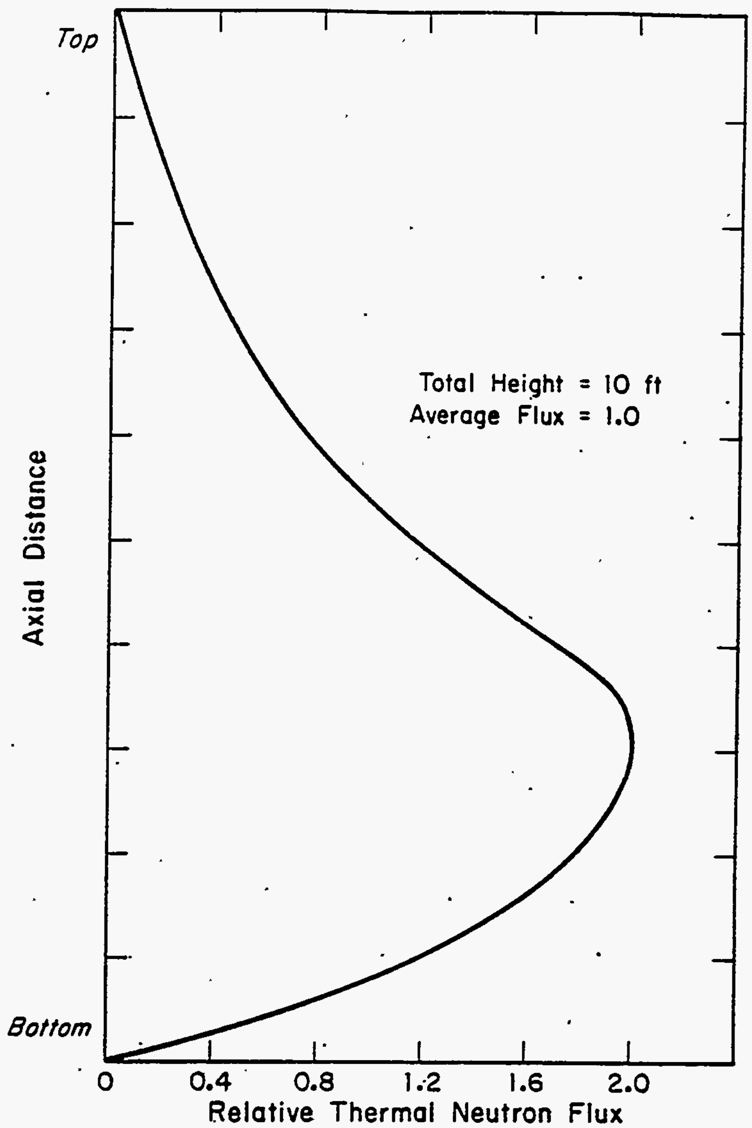

FIGURE 6. Axial Distribution of Thermal Neutron Flux

The content of each radioactive nuclide relative to the initial content of its precursor was calculated from the standard expression,

$$\frac{N_2}{(N_1)_{t=0}} = \frac{\alpha e^{-\alpha t}}{\beta} (1 - e^{-\beta t})\, e^{-\lambda_2 T}$$

where $\alpha = \phi\sigma_1$

$\beta = \lambda_2 + \phi(\sigma_2 - \sigma_1)$

$t =$ irradiation time

$T =$ decay time after shutdown

Nuclear data are given in Table 4.

TABLE 4

Nuclear Data for Induced Activity

Radioactive Nuclide				*Precursor*	
Name	*2200 m/sec σ_a, barns*	*$T_{\frac{1}{2}}$, yr*	*λ, sec^{-1}*	*Name*	*2200 m/sec σ_a, barns*
^{55}Fe	0	2.7	8.1×10^{-9}	^{54}Fe	2.3
^{60}Co	2	5.3	4.2×10^{-9}	^{59}Co	37
^{63}Ni	23	100	2.2×10^{-10}	^{62}Ni	14.2

The irradiation time used, 841 days, corresponds to the total period of HWCTR power operation. The activities were decayed from December 1964 to July 1975 (a period of 10 years and 7 months).

The equation for the ratio of the radioactive nuclide to its precursor was solved one time for each nuclide pair at a neutron flux of 10^{12} n/cm^2-sec. Relative contents in any reactor system component are proportional to those results multiplied by the ratio of the flux in that component to the 10^{12} flux.

304 stainless steel has the following nominal composition:

Element	*Content, wt %*
Cr	19.0
Mn	1.0
Fe	70.9
Co	0.1
Ni	9.0

The iron and nickel contents of any batch of stainless steel are carefully controlled and differ from the nominal fraction by ±10% or less; e.g., nickel comprises 8 to 10% of any batch of 304 stainless steel. However, ^{59}Co is an impurity and has no controlled lower limit. The ^{59}Co content of several samples of stainless steel analyzed at the Savannah River Laboratory (SRL) several years ago varied from 50 to 1100 ppm. A value of 1000 ppm (0.1%) was used throughout this evaluation.

Carbon steel contains only trace amounts of nickel and is comprised of about 99% Fe. The ^{59}Co content was assumed to be 1000 ppm.

The induced activity in a gram of stainless steel was calculated from the data given in Table 5.

TABLE 5

Specific Activities in Stainless Steel

Radioactive Nuclide	*Precursor Atom Fraction*	*Fraction of Element in SS*	N_2/N_1[a]	*Ci/g*	*Activity Content* *Ci/g SS*	*Fraction, %*
^{55}Fe	0.058	0.71	0.83×10^{-5}	2410	0.82×10^{-3}	49
^{60}Co	1.0	0.001	0.58×10^{-3}	1130	0.66×10^{-3}	40
^{63}Ni	0.036	0.09	0.95×10^{-3}	57	0.18×10^{-3}	11
				Total	1.66×10^{-3}	

a. For thermal neutron flux of 10^{12} n/cm^2-sec, an irradiation time of 841 days, and a decay time of 10½ years.

Radioactivity of Major Components

The residual activities of four major reactor components are given in Table 6. All components are made of stainless steel except for the reactor vessel, which is made of carbon steel with a thin (0.25 inch) stainless steel liner.

The thermal shield and monitor pin plate contain over 90% of all activity remaining at the HWCTR site.

Simple decay of the reactor components will result in significant reductions in the ^{55}Fe and ^{60}Co activities over 20 to 30 years (Figure 7). However the ^{63}Ni activity (100-year half-life) persists for a much longer period. The effects of postulated ^{63}Ni releases are evaluated (Consequence Analysis, page 62).

TABLE 6

Activity of Reactor Components

	Activity, Ci[a]			
	^{55}Fe	^{60}Co[b]	^{63}Ni	Total
Thermal Shield	10,800	8800	2400	22,000
Monitor Pin Plate	390	320	90	800
Reactor Vessel	380	230	10	620
Control Rods	100	80	20	200
	11,670	9430	2520	23,620

a. July 1, 1975.
b. From the 1000 ppm ^{59}Co impurity assumed in steels.

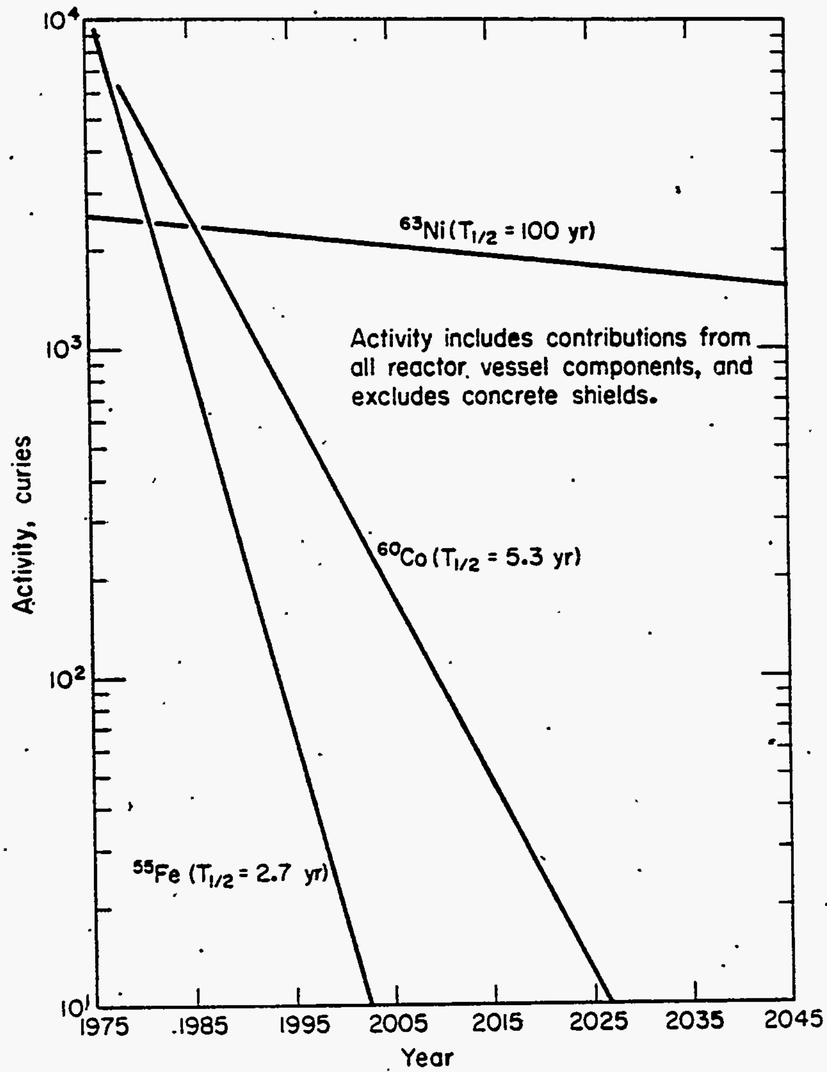

FIGURE 7. Activity Decay of Reactor Vessel Components

The highest specific activity in the system is found at the inside surface of the thermal shield, about 3 ft above the monitor pin plate. The activity there is 1.8×10^{-2} Ci/g of stainless steel, including all three radioactive nuclides. The axial average over the inner surface is one-half that value, or 0.9×10^{-2} Ci/g of stainless steel. The average value for the entire thermal shield is 0.18×10^{-2} Ci/g of stainless steel. The ^{63}Ni activity is 11% of the above activities.

Induced Activity in Concrete

The induced activity in concrete shields must be considered in any proposal for dismantling the HWCTR facilities. ^{60}Co is the radionuclide of most importance, originating from ^{59}Co that was present in steel shot or iron reinforcing rods, and also was present as a natural impurity of ordinary concrete. Three shield regions are of interest: the lower axial shield, the barytes-concrete annular shield around the lower portion of the vessel, and the biological shielding walls.

- Lower axial shield - a right circular cylinder, 3½ ft thick and about 5 ft in diameter. It consists of 90% steel shot by weight and 10% concrete. The total ^{60}Co activity of the shield is about 10 Ci, confined to the upper few inches of the shield nearest the reactor. Removal of the shield is necessary if the site were dismantled in the next 20-30 years.

- Barytes shield - an annular ring of concrete surrounding and extending above the axial shield. The iron content is 13% by weight. The total ^{60}Co activity of the shield is about 10 Ci. Removal of the shield is necessary if the site were dismantled in the next 20-30 years.

- Other concrete shielding - the remainder of the concrete shields is composed of ordinary concrete. No specifications exist for the ^{59}Co content of ordinary concrete. To obtain an estimate of the ^{59}Co content, two samples were obtained from the shield wall on the 37-ft elevation and were analyzed by neutron activation and by atomic absorption. Irradiated concrete is not accessible.

	^{59}*Co Content, ppm*	
	Sample 1	*Sample 2*
Neutron Activation	10	140
Wet Chemistry (atomic absorption)	44	130

The spread in results is not unexpected, considering the limited number of samples and the sample size. A value of 100 ppm was used in estimating the ^{60}Co activity in concrete near the reactor vessel at 400 pCi/g. The limit for ^{60}Co in concrete specified for the Elk River decommissioning was 0.04 pCi/g. At least 3 ft of concrete would have to be removed from inside the vessel cavity to achieve the Elk River limit. The reinforcing rods in the concrete would also have to be removed.

A better estimate of ^{60}Co activity in the concrete could be made by core drilling through the shield to the reactor vessel wall and analyzing the core material. If a dismantling plan is to be developed in further detail, these data would be useful in determining more precisely the concrete removal that would be necessary. The total volume would still be uncertain until the vessel cavity was exposed and some material actually removed. The 0.04 pCi/g value is a very stringent requirement. A severe cost overrun was experienced in the concrete removal costs in the Elk River dismantlement ($350,000 estimated versus $1.2 million spent).

Reactor Vessel Measurements

Direct radiation measurements on the reactor vessel were obtained to estimate personnel exposure rate and to confirm calculations of induced radioactivity in the reactor vessel. Measurements were taken in available openings through the biological shield including two power level sleeves and a neutron beam tube into the bottom of the vessel (Appendix A). Personnel dose rates are estimated to be 2-3 R/hr at a distance of 2 ft from the reactor vessel. Open process water lines near the reactor would increase dose rates by a factor of 100. Measurements through a neutron beam tube indicated 140 R/hr inside the bottom of the reactor vessel at the bottom of the monitor pin plate. Approximately 3 inches outside the bottom of the vessel, the radiation intensity in the tube was 7 R/hr.

The calculated specific activities of the thermal shield and reactor vessel (Table 6) were used to calculate the radiation levels at the three locations where data were obtained. The expression used in the conversion was for an infinite-slab source with slab shields located between the source and the detector. Source terms were expressed in ^{60}Co disintegrations/cc; flux terms at the detector were converted to mR/hr. The results are presented in Table 7.

Radiation levels outside the vessel are reduced by a factor of 10^3 because of the shielding afforded by the vessel and other intermediate materials. The relatively good agreement between calculated and measured results supports the neutron fluence (ϕt) and the ^{59}Co content of the thermal shield assumed in the calculations.

TABLE 7

Radiation from the Reactor Vessel

Location	*Radiation Levels, mR/hr* *Calculated*	*Measured*
SLV-1	490	500
SLV-4	120	210
Beam tubes (inside reactor vessel)	410,000	140,000

Radiation in Work Areas

Personnel radiation dose rates for general work related to disassembly and removal of equipment will be low with the exception of work associated with the reactor vessel. Exposure rates below zero level will average only 2-3 mR/hr and will be less than 1 mR/hr on zero level until the reactor tank top is removed or the reactor tank is exposed. Although available radiation data indicate dose rates of 2-3 R/hr 2 ft from the side of the reactor tank, it must be recognized that opening process water lines, removing the tank top or otherwise exposing the inside of the reactor will cause large increase in exposure dose rates.

Protective clothing required for work on zero level will be minimal (gloves, shoe covers, lab coats) except for work associated with the reactor tank and associated process water and off-gas piping. Initial line breaks on process lines will require assault masks or fresh air masks if burning is necessary. Work below zero level will require coveralls, gloves, and shoe covers. Fresh air masks will be required for burning or welding on contaminated equipment and may be required in some instances for opening process water lines for tritium protection. Survey records indicate that tritium-contaminated water has spilled from process line breaks in recent years. The maximum outgassing recorded is 40×10^{-5} μCi/cc of air during breaks in 1972.

The dome and some equipment in and above the zero level, including the crane, probably can be excessed for unrestricted use with a minimum of decontamination. This does not include process water, off-gas, and other equipment used in direct contact with reactor fuel, the tank, or the fuel storage basin.

Release for unrestricted use would require a detailed survey at the time the equipment is removed.

Radioactivity Guidelines

For each of the three decommissioning alternatives, activity guidelines are needed for direct personnel exposure, for indirect exposure via activity transport from the site, and for dispositions of materials removed from the HWCTR site. Documents cited below were reviewed to determine appropriate guidelines even though some of the regulations do not specifically apply to the HWCTR program.

The following abbreviations will be used for the regulations cited:

SRP-TS	- "SRP Technical Standard for the Release of Radioactivity from the Savannah River Plant," DPSTS-RH-W-0.1.
DPSOP-40	- SRP Operating Procedure DPSOP-40, "SRP Radiation and Contamination Control."
ERDA 5301	- Energy Research and Development Agency (ERDA) Manual Chapter 5301, Part VI, "Utilization and Disposal of Real Property."
PMI 109	- ERDA Property Management Instruction, Subpart 109-45.50.
RG 1.86	- Nuclear Regulatory Commission, "Termination of Operating Licenses for Nuclear Reactors," Regulatory Guide 1.86.
DOS	- Division of Operational Safety (DOS), draft, "Guidelines for the Safe Disposition of Contaminated Real or Related Personal Property."
Elk River	- "Experiences in Decontamination/Decommissioning of the Elk River Reactor." (D. McConnon and J. E. Menec, United Power Association, *Proceedings of the Second AEC Environmental Protection Conference*, WASH-1332 (74), p 785).

Specific values used in this decommissioning plan are given in Table 8.

TABLE 8

Radiation and Contamination Guidelines

Condition	Radiation or Contamination	Guideline Limit	Source of Limit	Applicable to Protective Confinement	Entombment	Dismantlement
Areas in Facility						
Clean Zone	βγ transferable	80 d/m	DPSOP-40	X		
	α transferable	10 d/m	DPSOP-40	X		
	^3H	50 c/m above background	DPSOP-40	X		
Regulated Zone	βγ, dose rate	300 mrad-50 mrem/hr	DPSOP-40	X		
Radiation Danger Zone	βγ, dose rate	>300 mrad-50 mrem/hr	DPSOP-40	X		
Materials Remaining in Facility						
Surfaces - Removable	Uranium α	1000 dpm/100 cm^2	RG 1.86		X	X
	TRU α	20 dpm/100 cm^2	RG 1.86		X	X
	βγ	1000 dpm/100 cm^2	RG 1.86		X	X
Surfaces - Fixed	Uranium α	5000 dpm/100 cm^2	RG 1.86		X	X
	TRU α	100 dpm/100 cm^2	RG 1.86		X	X
	βγ	5000 dpm/100 cm^2	RG 1.86		X	X
Dose Rate	βγ, unshielded, contact	0.3 mrem/hr	PMI 109		X	X
Concrete	^{60}Co	0.04 pCi/g	Elk River			X
Excess Equipment Removed from Facility						
Offplant - Unrestricted Use (Category A)						
Surfaces - Removable or Fixed	βγ	80 d/m	DPSOP-40	X	X	X
	α	10 d/m	DPSOP-40	X	X	X
Offplant - ERDA Contractors, etc. (Category B)						
Surfaces - Removable	βγ	10,000 c/m	DPSOP-40	X	X	X
	α	10,000 d/m	DPSOP-40	X	X	X
Surfaces - Fixed	βγ	500 mrem/hr (at 12")	DPSOP-40	X	X	X
	α	10^6 d/m	DPSOP-40	X	X	X
Onplant (Category C)						
Surfaces - Removable	βγ	10^5 c/m	DPSOP-40	X	X	X
	α	10^5 d/m	DPSOP-40	X	X	X
Surfaces - Fixed	βγ	5000 mrem/hr (at 12")	DPSOP-40	X	X	X
	α	10^7 d/m	DPSOP-40	X	X	X
Burial on Plant (Category D) (See Table 16)						

Dismantlement

For dismantlement, all equipment and structural materials containing or contaminated with radioactivity above guidelines are to be removed. The guidelines depend on whether or not the property will be reused for ERDA-controlled activities or released for uncontrolled public use (DOS). The proposed DOS guidelines include both RG 1.86 contamination limits, and limits for contaminated materials and equipment (PMI 109). Case-by-case review of decontamination criteria is also provided. A combination of the most suitable guidelines from the various sources is recommended for dismantlement. Adherence to these guidelines should permit release of the HWCTR site for uncontrolled use. Guideline limits were selected for contamination levels on materials remaining in the facility (RG 1.86, DOS); for dose rates from these materials (PMI 109), and for disposition limits for materials removed from the facility (DPSOP-40). In addition, 0.04 pCi/g of ^{60}Co in concrete is the upper limit for estimating the amounts of concrete to be removed. Elk River dismantlement used this limit for burial of concrete rubble in noncontaminated landfills. The proposed DOS guidelines also include the requirements for monitoring given in PMI 109. The ability to detect levels of beta-gamma radiation to 10 μrad/hr through not more than 7 mg/cm^2 absorber at 1 cm from the surface is required. Alpha detection capabilities are required to be 1000 dpm/100 cm^2 for nonplutonium alpha, and 100 dpm/100 cm^2 for plutonium alpha.

Entombment

For entombment, only some of the contaminated equipment external to the biological shield would be removed from the facility. The remaining radioactive or contaminated components would be sealed within a structure that would meet the criteria of prevention of access to the facility (DOS) and structural integrity over the period of time in which significant quantities of radioactivity remain with the material in the entombment (RG 1.86). A set of acceptable contamination levels are given in RG 1.86 (also proposed by DOS), and in the case of entombment these are defined as "significant quantities of radioactivity." Most of the radioactivity associated with the HWCTR is activation products induced throughout steel or concrete; therefore, the RG 1.86 standards (designed for decontamination criteria) need to be supplemented. A dose rate limit of 0.3 mrad/hr (PMI 109) was selected for exposure from equipment.

These guidelines are considered consistent with the proposed ERDA requirement for entombment: "Radiological safety criteria for the entombment of radioactivity will be developed on a case-by-case review and must be based on a hazards evaluation of the proposed action. These criteria must be approved by the Director, Division of Operational Safety (DOS)."

Protective Confinement

Protective confinement assumes ERDA control of the HWCTR site and the land containing ground-water migration pathways from the site to Upper Three Runs Creek. SRP regulations (DPSOP-40) that define clean, regulated and radiation danger zones would be applicable and adequate to prevent accidental exposure of personnel. The regulations also cover the disposition of contaminated equipment removed from the HWCTR site. Periodic surveillance would be required, and results should be well documented.

The present SRP Technical Standard for offsite effects of the release of radioactivity from SRP was selected as a conservative criteria to analyze the effects from radioactivity reaching the environs after HWCTR decommissioning. The primary reason for using this low dose is that decommissioning a radioactive facility involves securing the activity such that public effects are minimal. In fact, only a pessimistic combination of highly unlikely conditions without normal corrective action will initiate transport of the residual activity offsite. Therefore logic dictates that this activity transport once initiated from HWCTR activity will continue under the ground rules applied (Consequence Analysis, page 62).

The radionuclides remaining in the HWCTR in significant quantities are ^{60}Co (half-life 5.3 yr) and ^{63}Ni (half-life 100 yr). The properties of these nuclides and the estimated quantities are given in Tables 4 and 6. The concentrations in water that would result in the limiting doses were calculated for these two nuclides and for ^{239}Pu, which is present in the HWCTR but in very low quantities. If releases from the HWCTR reach an aquatic environment such as Upper Three Runs Creek or the Savannah River, radionuclidic concentrations in the food chain must also be considered. If both ^{63}Ni and ^{239}Pu are present in the drinking water, the limiting concentrations should be adjusted proportionally since they both contribute dose to bone. Calculated concentrations for isotopes of interest are shown in Table 9.

The derivation of the various concentration limits is given in Appendix B.

TABLE 9

Concentrations for 30 mRem/yr, μCi/cc

Nuclide	*Critical Organ*	*Drinking Water*	*Drinking Water and Eating Fish*
^{239}Pu	Bone	6.2×10^{-8}	4.4×10^{-8}
^{60}Co	GI Tract (Large Lower Intestine)	1.1×10^{-6}	7.4×10^{-7}
^{63}Ni	Bone	4.9×10^{-7}	1.3×10^{-7}

Other possible limits considered were from ERDA Manual Chapter 0524, "Standards for Radiation Protection and 10 CFR 100, "Reactor Site Criteria." The former considers effects from normal releases; limits in the latter regulation pertain to reactor accidents. All limits are compared in Table 10 including the recent radiation dose standards for the uranium fuel cycle proposed by the Environmental Protection Agency (EPA).

The SRP Technical Standards were selected for the following reasons:

- Present SRP operations are conducted within these limits. Predicted releases from an inactive facility should be lower even for consequences that result from unexpected conditions many years in the future.

- It is probable that the ERDA 0524 guides will be lowered in the future, particularly if the EPA's proposals are accepted.

- The 10 CFR 100 guides pertain to major accidents that have a potential for large releases of radioactivity. No such situation exists in HWCTR.

- Other decommissioning operations also used standards that apply to normal operations (10 CFR 20 for Hallam, Bonus, etc); Elk River used Appendix I to 10 CFR 50.

TABLE 10

Dose Limit Comparisons - Public Zone

Type of Exposure	SRP Tech Std, mrem/yr Individual Max	ERDAM 0524, mrem/yr Ind Max	Pop Avg	EPA Proposal, U Fuel Cycle, mrem/yr	10 CFR 100, mrem
Whole body	10	500	170	25	25,000
Gonads	10	500	170	25	-
Bone marrow	10	500	170	25	-
Gastrointestinal tract	30	1500	500	25	-
Bone	30	1500	500	25	-
Thyroid	30	1500	500	75	300,000
All other organs	30	1500	500	25	-

CRITERIA

Criteria were developed to weigh the alternatives of decommissioning HWCTR. The criteria include special factors that may be unique for the Savannah River Plant site but in general reflect guidelines proposed for ERDA facilities. The criteria are radiation impact, cost, land area commitment, timing, and aesthetics.

Radiation Impact

The radiation impact of decommissioning HWCTR consists of several elements. In the completion of decommissioning, the work force will be exposed to an occupational dose. Estimates of the dose are made from radiation surveys and Construction Division estimates. After decommissioning, two potential paths for additional dose to the public exist via exposure to direct radiation from the vessel or ingestion of water contaminated by the vessel. The Safety Analysis section includes the evaluations of the likelihood and consequences of radiation from HWCTR activity for all alternatives.

Cost

The cost elements in each alternative are the capital cost of the decommissioning step and the operating cost for surveillance, maintenance, and monitoring. Lowest cost alternatives that protect the public should be favored because tax money will be spent for decommissioning.

Land Area Commitment

The land area associated with each alternative is estimated from current site maps. The acreage in U area is already a small portion of the SRP site, but it also may be required to retain the water rights between U area and Upper Three Runs Creek for all cases except dismantling the reactor. The importance of land depends heavily on the long-term ERDA plans for the site (currently undefined).

Timing

This factor is a judgment of what decommissioning action (and when) is in the best interest of the taxpayer. The effort will be simplified by ^{60}Co decay in 35-70 years, and there is no immediate incentive to act as long as the activity is secure in the interim. On the SRP site, such security is essentially guaranteed. However, there is always some merit in finally resolving a decommissioning step rather than continuing studies under changing rules.

Aesthetics

Although less significant than other criteria, some of the public prefer an approach that restores the land to preconstruction state. The emphasis of aesthetics (as with land area) would differ for a more visible reactor site rather than an isolated region as at the SRP site. Industrial reactor sites or sites containing multiple reactors should be weighted differently on aesthetics than a site reusable for industrial applications.

ALTERNATIVES

Three alternatives for decommissioning HWCTR were evaluated in developing this plan:

Alternative	*Objective*
Dismantlement	Restore the U site to a condition suitable for release to the general public. Relocate HWCTR radioactivity to the site burial ground.
Entombment	Secure the activity remaining in the HWCTR facility so that release to the environment is extremely improbable until decay renders it harmless.
Protective Confinement	Confine activity at the HWCTR site in dry storage for the forseeable future while decay proceeds.

Alternatives selected for decommissioning HWCTR were in part derived from a review of decommissioning action of other similar reactors. As shown in Table 11, only one reactor (Elk River) has been dismantled (1973) both as an AEC demonstration and to meet the political requirement of no residual activity in the state of Minnesota. Parts of the reactor are now in commercial burial grounds in Illinois, Kentucky, and Washington. Four reactors have been entombed in varying degrees; Bonus entombment is the most similar to the HWCTR case studies. Six* reactors are now decommissioned in a state similar to protective confinement. In all of these cases (including Elk River), the utility owns the land site and in some cases is operating other reactors nearby.

Within each broad alternative, options were evaluated as separable cost items for consideration in the scope of work for each case. Different methods of meeting the objective are also recorded without feasibility or costs for future consideration as part of the discussion of alternatives.

Key features in alternatives for decommissioning HWCTR are compared in Table 12.

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* Proposed for Peach Bottom 1 subject to NRC approval.

TABLE 11

Decommissioned Reactors

	Type	Power Level, MW		Operating Interval	Decommis-sioning Interval	Operating Exposure, KMWD	Approx. Activity Inventory, curies	Cost, millions of dollars
		Thermal	Electrical					
HWCTR	HWR	50	--	1962-1964	--	14	10^4 (1975)	
A. Dismantled/Land Released								
Elk River	BWR	58	22	1964-1968	1972-1974	53	10^4 (1971)	~6
B. Entombment - Release of Facility								
Hallam	Sodium	240	75	1963-1964	1969	--	--	~3
Entombment of Nuclear Section - Use of Support Facilities*								
Bonus	BWR	50	16.5	1964-1968	1970	10	5×10^4 (1970)	--
Piqua	OMR	45	11.4	1964-1966	1968	~15	5×10^4 (1969)	--
AFNEC	Test	10	--	1965-1970	1971	3	10^5 (at shut-down)	0.5
C. Mothballed								
Saxton	PWR	28	3	1962-1972	--	--	10^5 (at shut-down)	--
CVTR	PHWR	65	17	1963-1967	1967	--	--	--
Peach Bottom - 1	HTGR	115	40	1966-1972	Now underway	--	10^6 (6 mos. decay)	--
Pathfinder**	BWR	190	58.5	1964-1967	1968	--	--	--
EBR - 1	Sodium	--	.15	1951-1964	--	--	--	--
Fermi	Sodium	200		1965-1966 1970-1972	1973 - 1974	6	5×10^3 (6/1/73)	7***

*Support facilities used for storage, for support facilities to other reactors on site, etc.

**Converted to oil-fired station.

***<5 without fuel reprocessing.

TABLE 12

Comparison of Alternatives

	Cost, $thousands				Land Area Committed, acres				
Alternative	Capital Project[a]	Burial Ground	Surveillance Operating 1975-2000	Total	Burial Ground	HWCTR Site	Water Rights Retained	Occupational Dose, rem	Aesthetic
A-1 Dismantlement	5400	120	0	5520	2	0	0	20	Best
A-2 Effect of capping building at zero level instead of -16	4400	120	0	4520	2	0	0	20	Good
B-1 Entombment	1700	20	~30	1750	0.2	1.5	0-90	5	Good
B-2 Effect of filling building with concrete	1600	20	~30	1650	0.2	1.5	0-90	5	Good
C Protective Confinement	190	-	~60	250	0	1.5	90	<1	Least attractive

a. Engineering Department Current Assessment of Cost for Evaluation, not Budget Quality and based on this schedule:

Alternative	Authorization	Construction Start	Construction Complete
A	1/78	8/78	7/79
B	1/78	3/78	10/78
C	1/78	4/78	8/78

Dismantlement

The objective of dismantlement is to remove all of the equipment with residual radioactivity above specified levels, so that the site could be released to the general public without restrictions. The residual radioactivity guidelines are discussed in *Radioactivity Guidelines* (Table 8). The practical implication of these guidelines is to require all of the equipment that was normally in contact with D_2O be removed. About 1300 ft^3 of radioactive reinforced concrete around the reactor cavity, the lower axial shield, and portions of the spent fuel basin liner are to be removed. In Tables 13 and 14, equipment to be removed and buried is specified.

The containment dome (Figure 8) and other above-grade structure of buildings are to be dismantled and removed from the site with no credit for sale or reuse. The steel and concrete structure of the reactor building (Figures 9 and 10) is to be removed to a depth of about 16 ft below grade, the remaining building cavity backfilled and capped with a concrete pad at the 16-ft depth, and the remaining cavity backfilled to grade level to clear the site for future construction.

The physical volume of HWCTR equipment to be removed and buried is about 14,000 ft^3. The estimated burial ground area required is 2 acres with normal burial practice, or about 1% of the total burial ground site. The cost of burying the equipment is estimated at $120,000. About 3400 ft^3 of equipment, including the reactor vessel, would require burial in the high level trenches (Table 13), and the balance in the low level trenches. About 100 ft^3 of equipment might require burial in the alpha trenches.

TABLE 13

Dismantlement Alternative - Equipment to be Removed and Buried in High Level Trenches[a]

EP No.	*Description*	*Location*	*Size*	*Volume, ft^3*	*Weight, ton*
1	Reactor Vessel with Internals	-	37' × 10'5" × 7'9"	3000	98
1	Lower Axial Shield (W231697)	1' Below Reactor Vessel	58"D × 37"	57	6
-	Boiling Loop Bayonet Stubs (4" SS Pipe)	-16'	About 30' of 4" Pipe	4	
-	Reactor Nozzles		About 24' of 10" Pipe	20	
-	Reinforced Concrete	Biological Shield		~500	

a. Category B equipment, beta-gamma high level waste (Table 16).

TABLE 14

Qismantlement Alternative - Equipment to be Removed and Buried in Low Level Trenches[a]

EP No.	Description	Location	Size[b]	Volume, ft^3	Weight, lb[c]
41	Main Storage Tank	-52'	20'6" × 9' × 9'	1660	
20.1	Steam Generator	-16', -37'	23'3" × 6'1" × 6'1"	860	37,800
20.2	Steam Generator	-16', -37'	23'3" × 6'1" × 6'1"	860	37,800
21.1	Main Pump Without Motor	-16'	12'8" × 7'6" × 6'1"	546	
21.2	Main Pump Without Motor	-16'	12'8" × 7'6" × 6'1"	546	
86	Gas Recompressor	-52'	8'6" × 5' × 4'	170	
86.1	Gas Recompressor	-52'	8'6" × 5' × 4'	170	
194	ICL[d] Storage Tank	-52	8' × 4' × 4'	128	
42	Makeup Pump Without Motor	-52'	7' × 5'9" × 2'10"	113	3000
103	SFB[e] Deionizer	-52'	5'7" × 4'4" × 4'4"	105	24,000[f]
104	SFB Filter	-52'	5'7" × 4'4" × 4'4"	105	24,000[f]
186.1	LL[g] Pump Without Motor	-16'	7'1" × 4'4" × 3'4"	102	
186.2	LL Pump Without Motor	-16'	7'1" × 4' × 4" × 3'4"	102	
53	Hold Tank	-52'	11'8" × 21" × 21"	36	
101.1	SFB Cooler	-52'	13' × 26" × 13"	30	2800
101.2	SFB Cooler	-52'	13' × 26" × 13"	30	2800
51	Drain Tank	-52'	4' × 32" × 32"	28	
47	Collection Tank	-37'	5'1" × 26" × 26"	24	250
92	Catch Pot	-52'	46" × 30" × 30"	24	
180	ICL Seal Pump	-52'	35" × 29" × 38"	22	
195	ICL Seal Pump	-52'	35" × 29" × 38"	22	
193	ICL Afterfilter	-37'	4' × 2'7" × 2'	21	
45	Main Afterfilter	-37'	3'5" × 2' × 2'	14	
191	LL Purge Cooler	-37'	32" × 23" × 25"	11	

(continued)

TABLE 14 (Continued)

EP No.	Description	Location	Size[b]	Volume, ft^3	Weight, lb[c]
-	Four Process Valves	-16', -37'	4'8" × 3' × 2' ea	28 ea	
-	One Process Valve	-16'	3' × 2' × 2'	12	
-	10" Process Piping	-16', -37'	About 250'	>180[h]	
-	4" Process Piping	-16', 37', -52'	About 1000'	>110[h]	
-	12" and 8" Vent Piping	0'	About 100'	>60[h]	
22	Seal Head Tank	+52'	5' × 30" × 26" (5' × 20"OD)	27	1075
84	Vent Condenser	0'	7'10" × ~14" × 14" (4'7" × 8" D)	10	200
84.1	Separator	0'	5'2" × 14" × 14" (4'2" × 10" D)	7	344
181	ICL Seal Hold Tank	+52'	4' × 24" × 24"	16	
198	ICL Seal Hold Tank	+52'	4'24" × 24"	16	
60	Poison Tank	+52'	6'9" × 2' × 2'	27	
913.01	Tank	+52'	3' × 24" × 24"	12	
270	Transfer Coffin		25'4" × 2'8" D	192	
	Transfer Coffin Platform	0'	6'8" × 7'4" × 2'	100	
256	Rod Drive Platform	0'	8' × 14'6" × 9'4"	1090	
-	Reinforced Concrete	Biological Shield	-	800	
-	Eight Failed Fuel Element Containers	Top of SFB	12"D × 12'	8 ea.[i]	
-	Component Receptacle	Top of SFB	12' × 2' × 2'	48[i]	
178.1	BL[j] Pump Without Motor	-37'	7'1" × 4'4" × 3'4"	102	
178.2	BL Pump Without Motor	-37'	7'1" × 4'4" × 3'4"	102	
44.1	Main System Deionizer	-37'	5'8" × 4'8" × 3'8"	97	
44.2	Main System Deionizer	-37'	5'8" × 4'8" × 3'8"	97	
40.1	Main Purge Cooler	-37'	19' × 2' × 2'6"	95	
40.2	Main Purge Cooler	-37'	19' × 2' × 2'6"	95	
43	Seal Pot	-37'	13' × 2'8" × 2'8"	92	
105	Hold Tank	-16'	9'2" × 3' × 3'	83	
187	LL Cooler	-16'	12'10" × 2'2" × 2'2"	52	
54	Main System Prefilter	-37'	6' × 3' × 3'	54	{1150 Steel 7600 Lead

a. Category A equipment, beta-gamma low level waste (Table 16).
b. Size of equipment includes all projecting nozzles, support legs, etc.
c. Weight of equipment is given when unusual or readily available.
d. ICL - Isolated Coolant Loop.
e. SFB - Spent Fuel Basin.
f. With lead-shielded cask.
g. LL - Liquid Loop.
h. Volume of straight pipe, no bends included.
i. These could have internal alpha contamination and disposal in alpha trench could be required.
j. BL - Boiling Loop.

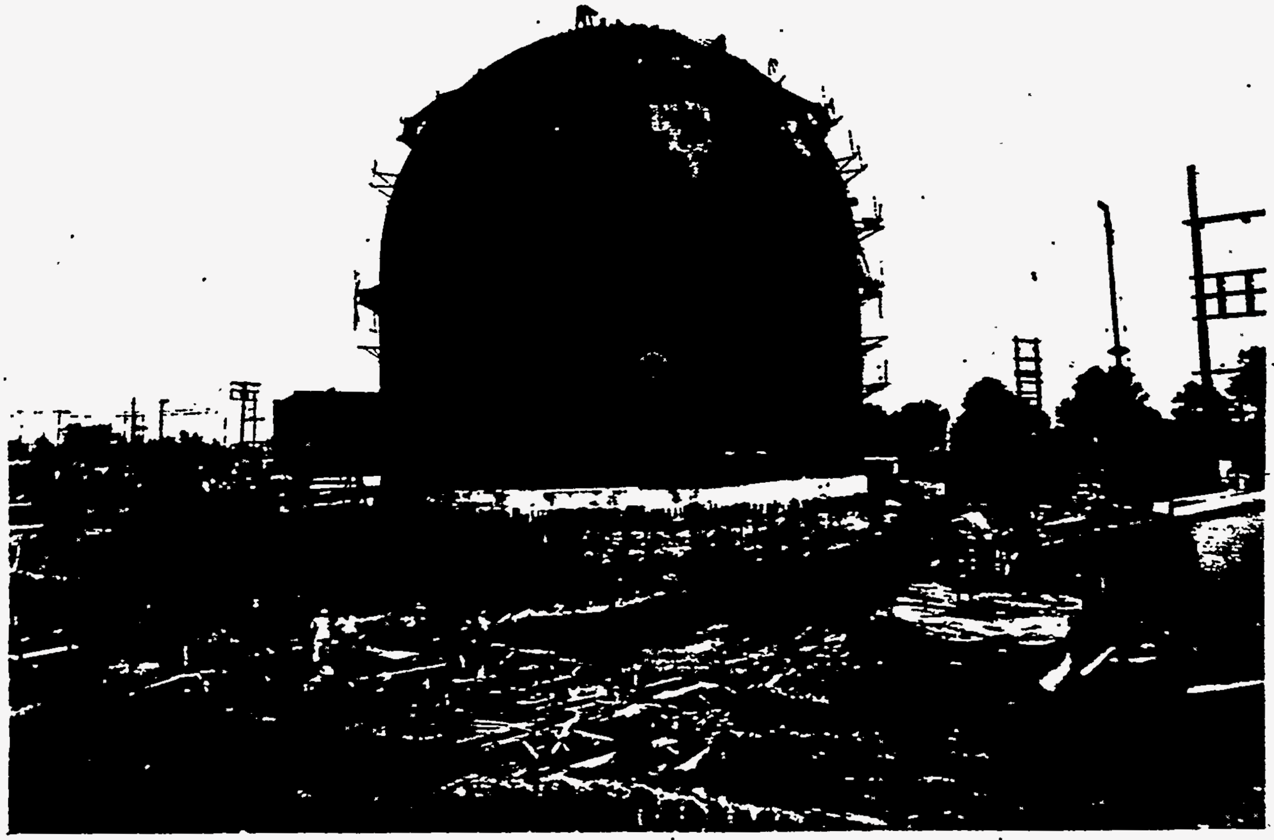


FIGURE 8. Construction of Above-Grade Steel Shell

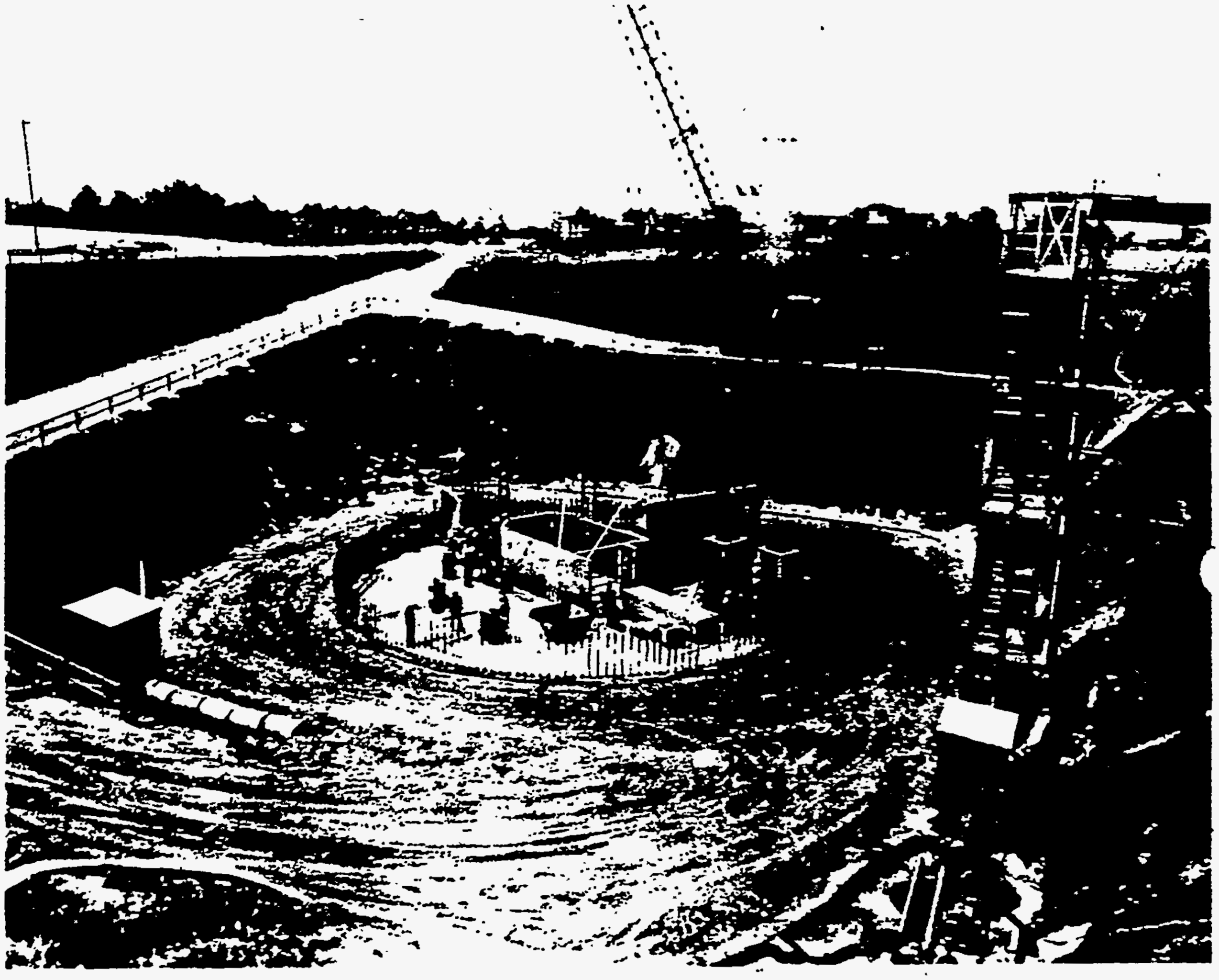


FIGURE 9. HWCTR Base Slab

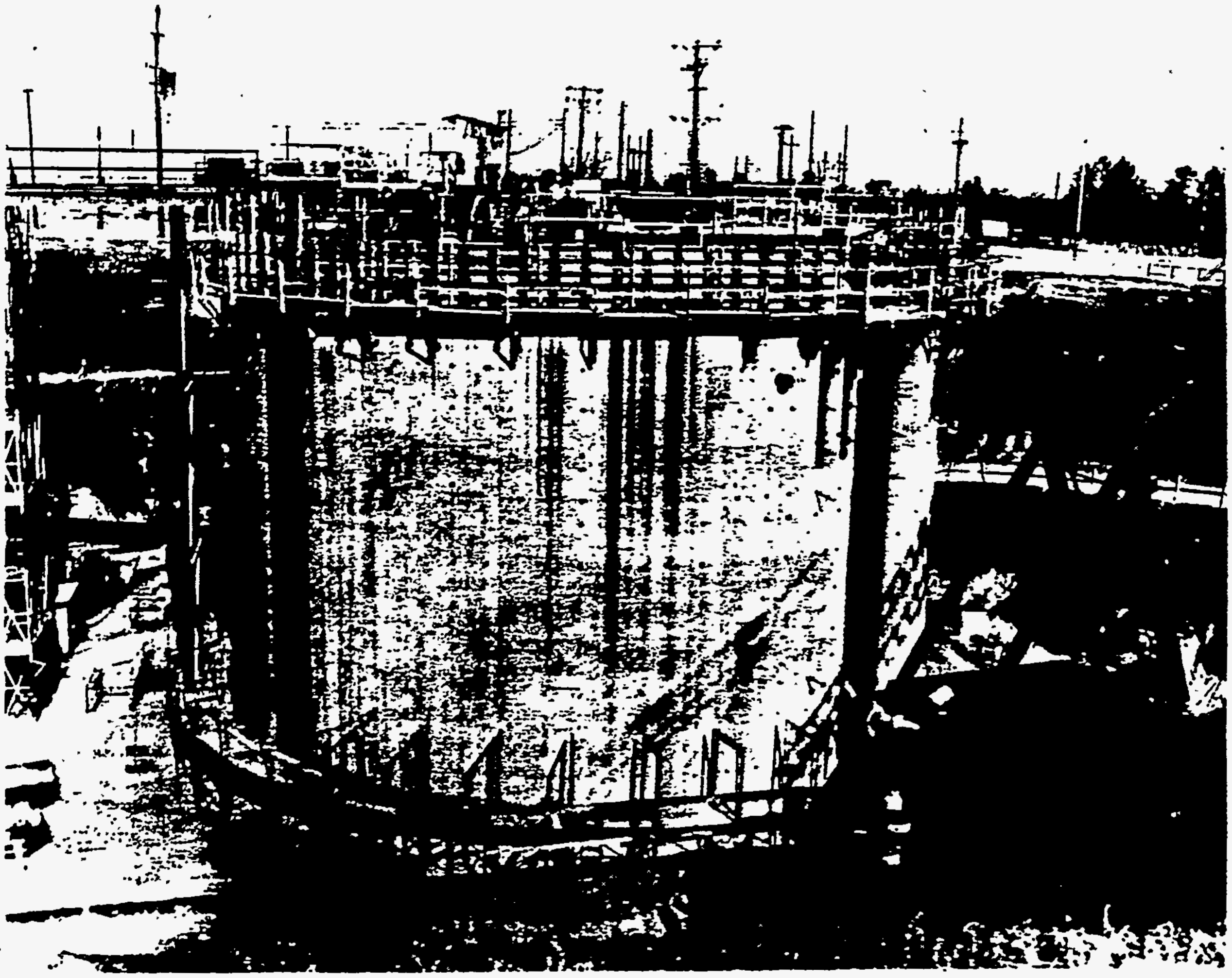


FIGURE 10. HWCTR Concrete Shell During Construction

The most significant engineering problem is removal of the reactor pressure vessel, which weighs 100 tons and contains the highly radioactive internal thermal shield plates. As discussed on page 25, the radiation level is only about 2-3 R/hr outside the reactor vessel but about 200 R/hr inside the vessel. Because the thick pressure vessel wall (3½-4½ inches of carbon steel) provides shielding from the highly radioactive internal parts, removing the reactor vessel and internal parts intact would be desirable and less costly than cutting the vessel into smaller pieces (as was done at Elk River). A crawler crane of sufficient lift capacity is available (Figure 11), and relocating the vessel to the onsite burial ground about 5 to 6 miles away via plant roads entirely within the plant security fence is considered feasible. Removal of the reactor vessel involves the following steps (Figure 12):

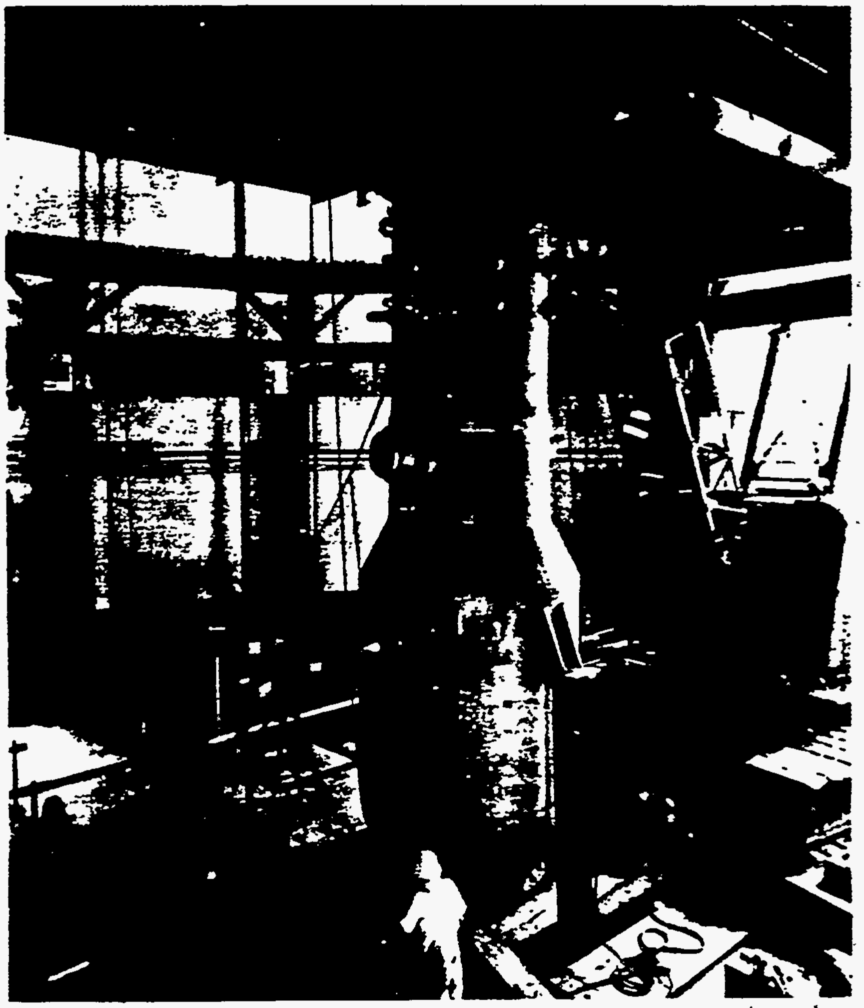
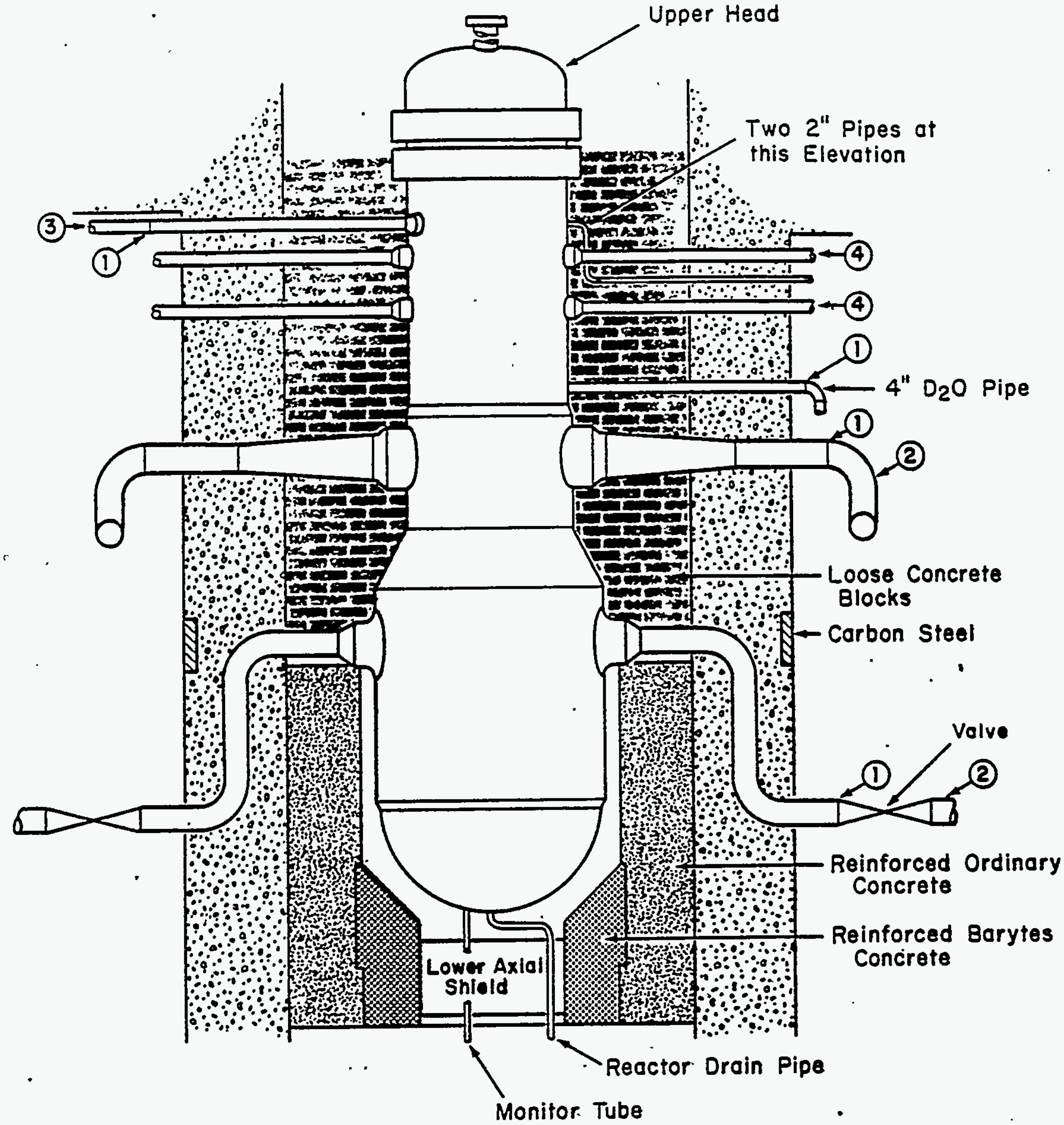


FIGURE 11. Reactor Vessel Being Moved Into Position



① Carbon Steel to Stainless Steel Transition

② 10" Carbon Steel Pipe

③ Two 4" Instrument Pipes at this Elevation

④ Six 4" ICL Pipes at this Elevation (Stainless Steel)

FIGURE 12. Reactor Vessel and Concrete Biological Shielding to Be Removed

- Disconnect control and safety rod housings from the vessel upper head, remove rod drive platform with housings, and seal upper head nozzles with blind flanges.
- Disconnect all monitor tube and thermocouple connections from the monitor pin sleeves beneath the lower axial shield, and seal the monitor pin sleeves with fittings that will allow the sleeves to be lifted up through the lower axial shield.
- Remove the loose concrete blocks from the cavity between the vessel neck and the poured concrete.
- Cut all the pipes that protrude radially from the pressure vessel and enter the concrete biological shield: four 10-inch pipes, seventeen 4-inch pipes, and four 2-inch pipes.
- Lift the pressure vessel out of the cavity and place it on a transport vehicle. The 2-3 R/hr general radiation level 2 ft from the core region of the vessel will not permit extensive work near the reactor but will not require elaborate shielding for crane operators or transport vehicle drivers. However, the 10-inch nozzle openings will emit two high radiation beams from the internal shield. Temporary shielding will probably be installed in these two openings (e.g. lead plugs).

Measures that reduce corrosion of the reactor structure will minimize the release of radioactive corrosion products to the ground water. For this reason, the reactor vessel penetrations (cutoff nozzles) must be sealed with watertight and corrosion-resistant seals to prevent water penetration. The corrosion resistance of the seals should equal that of the minimum thickness of stainless steel, i.e., 0.2-inch wall thickness of the bottom drain pipe. Additional barriers to water penetration would be provided. For costing purposes, the reactor vessel is assumed to be placed in a steel-lined concrete vault. Other methods, such as casting the reactor in concrete would be considered as part of the final design if this alternative is selected.

Removing the reactor vessel as a unit is considered to be much better than cutting up the vessel and internals for removal in smaller pieces. The cutting operation would release airborne activity (from torch cutting), which would require a confinement and air filtration system and breathing air protection for personnel and could interfere with other dismantling operation. Handling of the highly radioactive reactor internal parts would require personnel shielding systems and shielded transport casks and would almost certainly result in increased personnel exposure and increased risk of accidental exposure. In addition, the reactor internal parts would be buried in containers with corrosion resistance at least equivalent to that of the reactor pressure vessel (minimum 0.2 inch of stainless steel).

In the Elk River dismantlement program, the reactor vessel was cut up into pieces for removal; however, the radioactivity level and transport conditions were very much different from those of the HWCTR. First, the radiation level of the outer surface of the reactor vessel was 30 R/hr versus 2-3 R/hr for the HWCTR vessel. Second, the disposal plan required transport of the radioactive equipment out of the state of Minnesota. Transport of the large (7 ft diameter × 25 ft length), highly radioactive vessel for long distances on public roads or railroads involved a number of technical and regulatory questions that encouraged cutting up the vessel. In particular, a special shipment of a reactor vessel would require a very large heavy cask to lower the radiation level to 10 mR/hr at 6 ft and also several months of delay for approvals. In addition, a commercial burial ground would require a special review and approval by appropriate Federal and State agencies to receive such an unusual shipment.

The lower axial shield (Figure 13) directly under the reactor vessel could be removed by lifting it straight up through the empty reactor cavity, provided that the grout or packing in the 1-inch annular space between the shield and the surrounding concrete does not prevent upward motion. The shield (Appendix C) weighs about 6 tons, and the upper plate is activated to about 100-200 mR/hr. If the grout must be removed or loosened, access to the bottom of the annular space could be attained by removing the shield support lugs that are bolted to the pin room ceiling. Access to the top of the annular space could be attained by installing temporary shielding on top of the shield and, if necessary, around the sides of the cavity.

Portions of the concrete biological shield must also be removed. Calculations indicate that the barytes concrete at the bottom of the reactor cavity and the 2½-ft-thick annular section around the core region must be removed as shown in Figure 12. The amount of reinforced concrete to be removed (1300 ft^3) is an estimate because the ^{59}Co content varies widely. As part of the final engineering design of dismantlement, the estimated amount of concrete to be removed should be refined by taking core samples. A good estimate is important because the concrete removal could be costly in terms of dollars and radiation exposure. At Elk River, the cost of dismantling the concrete biological shield was large and was underestimated (about \$1.2 million versus \$0.35 million estimate). The reinforcing steel and the confined area interfered with dismantlement and required the use of explosives. Removal of the process piping from the concrete around the reactor would require removal of some additional reinforced concrete (Figure 12). This additional concrete would probably not have enough radioactivity to require burial.

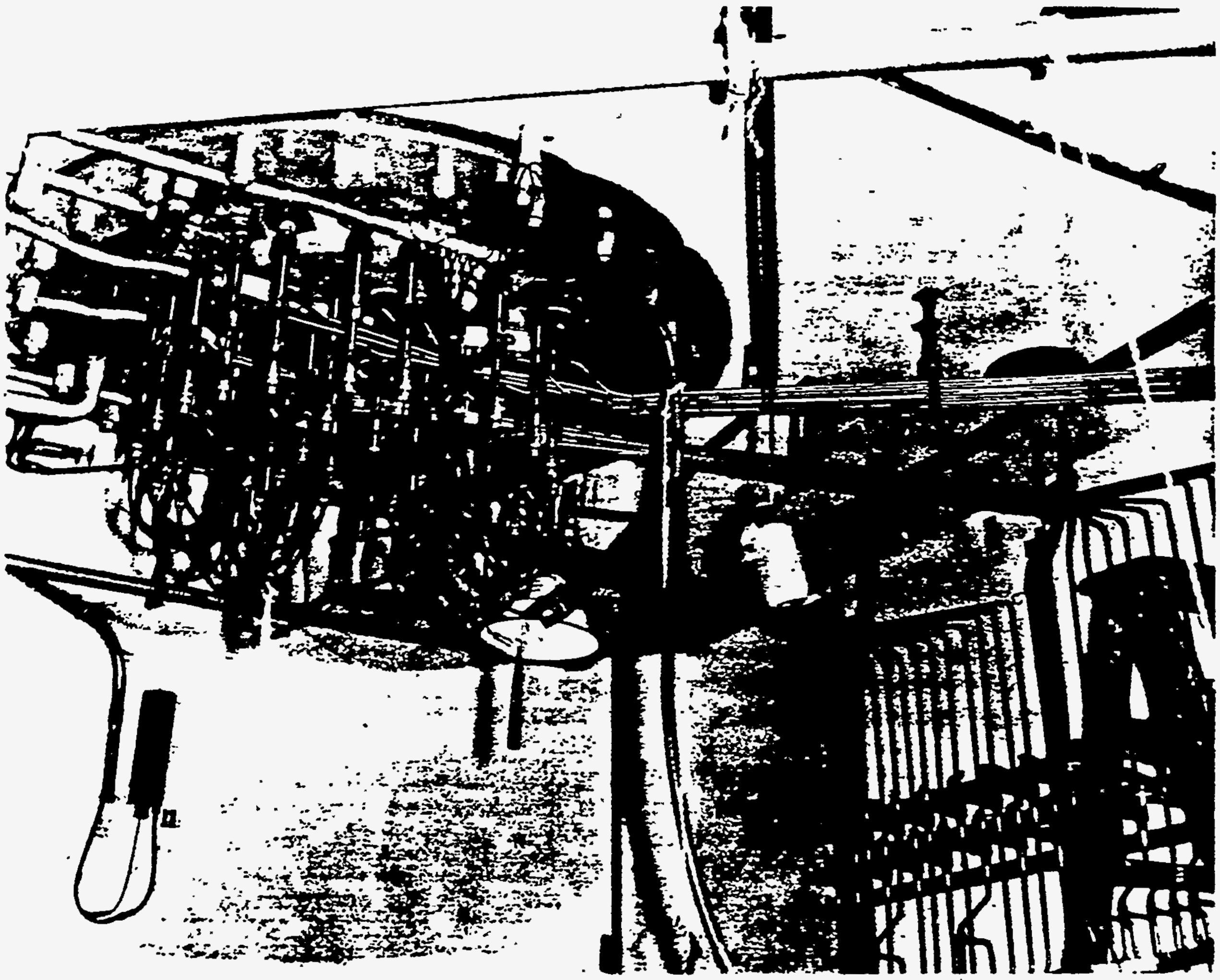


FIGURE 13. Bottom of Lower Axial Shield Showing Monitor Tubes and Segmented Support Rings

The cost saving for simply capping the building cavity at grade level rather than removing all material to a depth of 16 ft is estimated at one million dollars. This saving could be considered if the above-grade concrete cap is not detrimental to the future site use.

Entombment

The objectives of entombment of HWCTR are 1) to provide long-term (100 years or longer) security for residual radioactivity and thereby minimize the risk to the public, and 2) to minimize required maintenance and surveillance of the HWCTR site.

Two configurations were considered in the conceptual design of entombment structures.

Basic Entombment (Figure 14)

- Remove all above-grade contaminated equipment (Figures 15, 16, and 17) and piping. Place below grade.
- Remove actuator structure, leave head on reactor, and seal reactor head nozzles with blind flanges.
- Seal all concrete penetrations.
- Remove all remaining above-grade structures including steel dome and 25-ton crane for salvage or disposal.
- Cut pipes from pressure vessel at stainless steel/carbon steel transition and weld stainless steel plug in lines, minimum of 0.2-inch thick or equivalent plug.
- Backfill building below grade with compacted earth.
- Pour a reinforced concrete pad, approximately 1-ft thick over entire grade elevation of reactor structure. The area around the reactor head is to be approximately 3-ft thick.
- Cover the entire concrete surface with waterproof barrier. Cover with clay sloped for drainage. Seal with waterproof membrane. Install drain field to remove rainfall runoff from area.

Solid Entombment

- Include all items of basic entombment above except backfill.
- Prior to installation of roof slab, fill entire below-grade structure with concrete instead of earth.

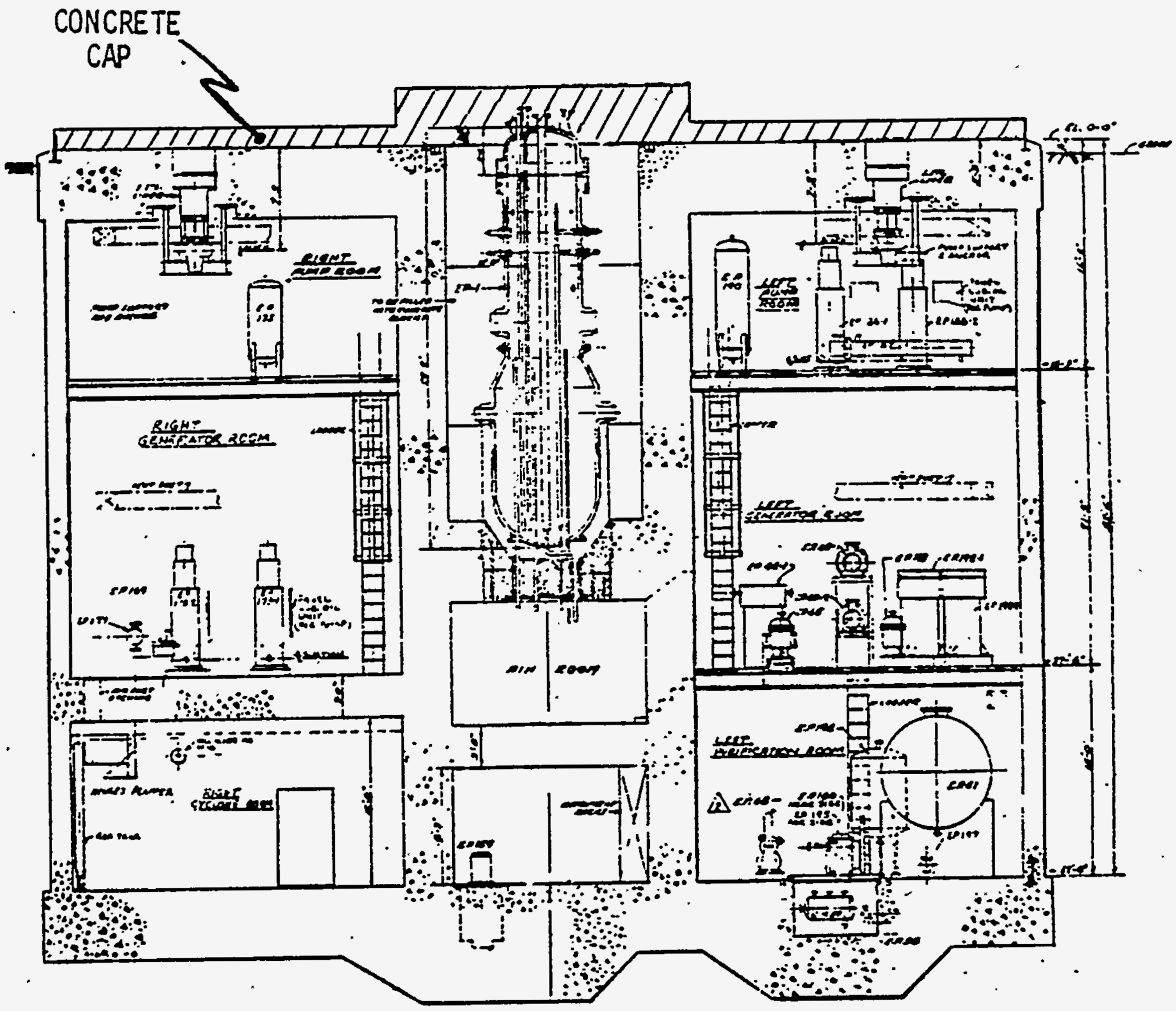


FIGURE 14. Basic Entombment

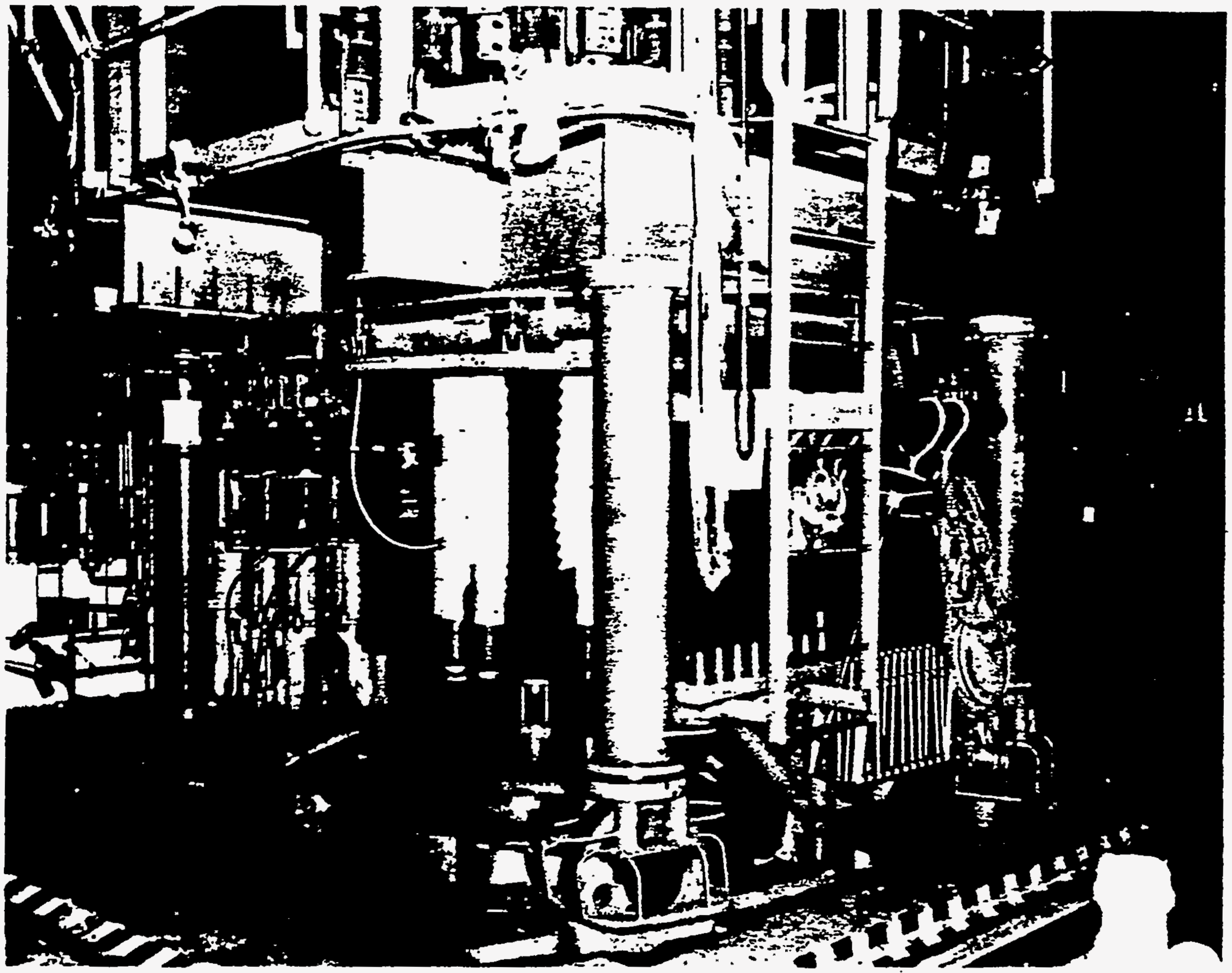


FIGURE 15. Reactor Head with Control Rod Drives

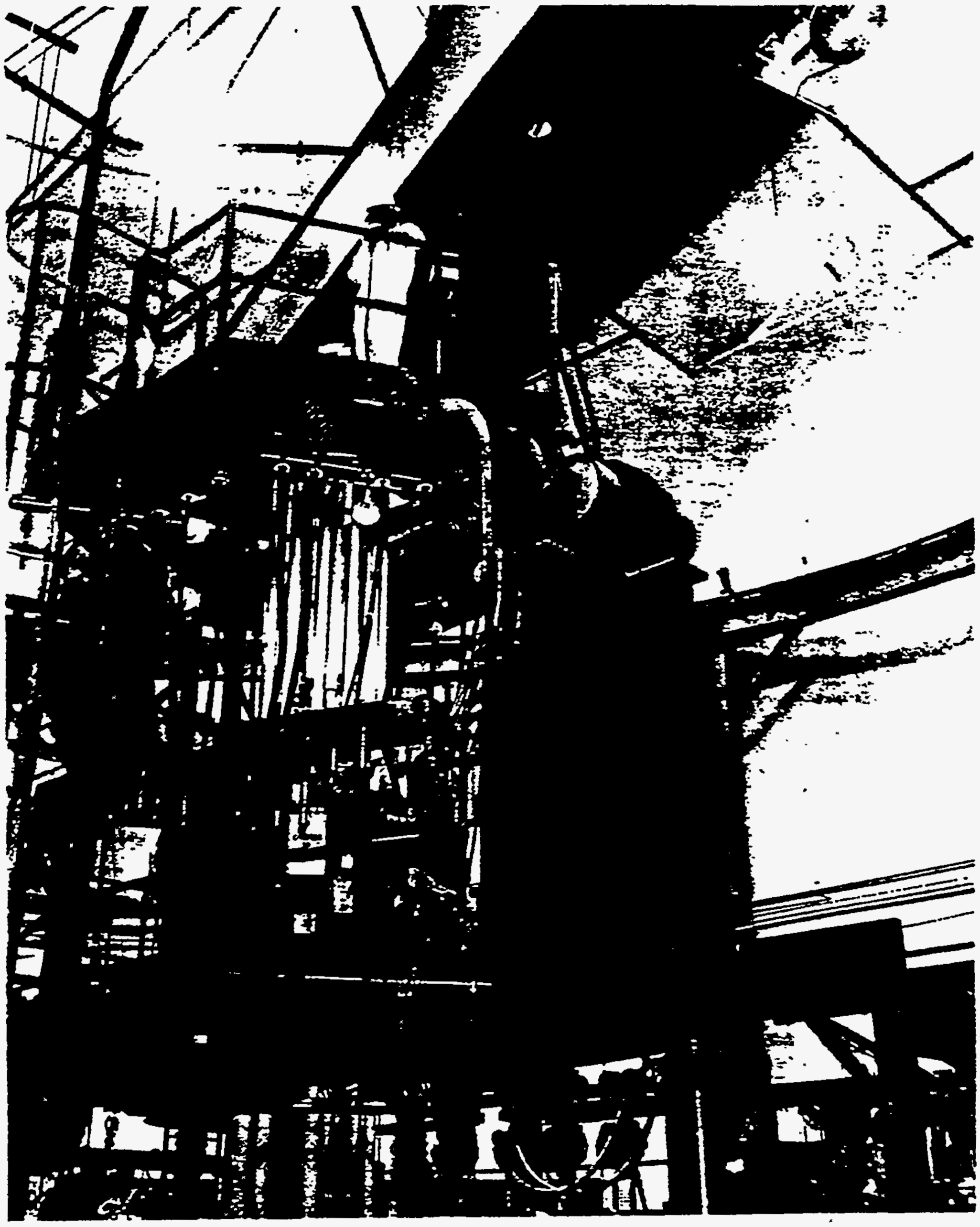


FIGURE 16. Control Rod Drive Platform

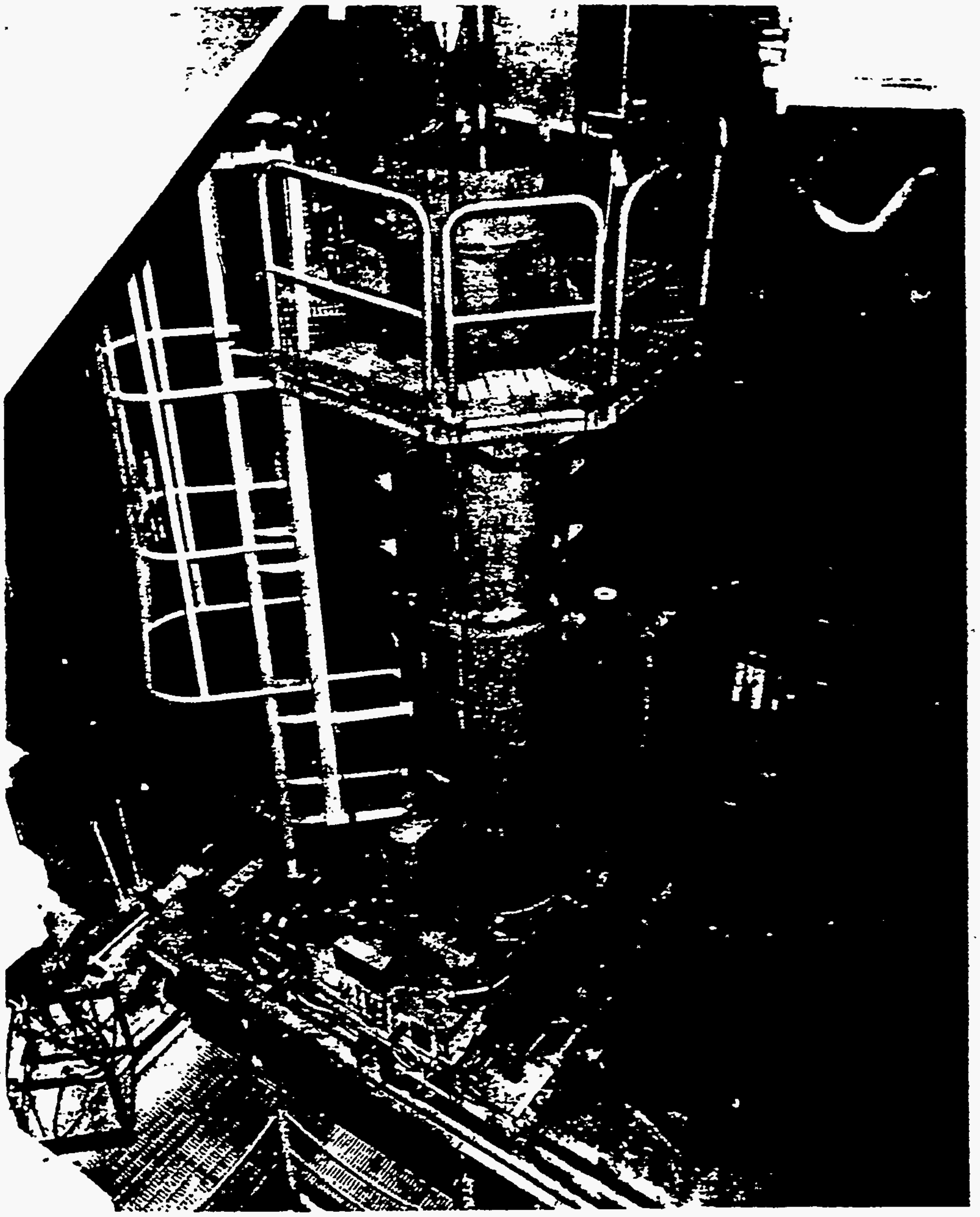


FIGURE 17. Fuel Transfer Coffin

Two additional configurations were considered:

1. Fill central core of below-grade building around pressure vessel, in pin room and monitor room with concrete. Encase the reactor vessel in a continuous cylinder of concrete from the foundation to grade level. Tying the pressure vessel to the foundation in this continuous cylinder would increase the resistance to earthquakes. An earthquake analysis of the structure will be needed to determine if this is necessary. This item has not been costed by the Engineering Department.

2. Use the present steel dome and concrete substructure as the entombment structure. All exterior doors, hatches, or openings would be permanently sealed by welding and other methods. The dome would be stripped of insulation, sand blasted, and painted as described in the protective confinement proposal. The concept was discarded for several reasons. Physically it is similar to protective confinement except that below-grade periodic inspection is not very convenient. Reduced surveillance would decrease annual expenditures, but there is no advantage over protective confinement because:

   - Physical security of the entombed radioactivity is not substantially improved over protective confinement.

   - Periodic inspection allowed by protective confinement would detect defects such as any increased water seepage into the concrete substructure.

   - The same yearly maintenance would be required for domed entombment as for protective confinement. The dome would require periodic exterior painting in either case.

Protective Confinement

The objective of protective confinement is to maintain the HWCTR equipment in a state of dry storage, with lower capital expenditure than that of entombment or dismantlement. The equipment would be left in place, but steps would be taken to prevent access of water to the reactor. Physical security for the radioactive inventory to prevent public exposure would be ensured by the current state of restricted access to the SRP site plus permissive access to the locked HWCTR area. The scope of work is as follows:

- Install flanges or welded seals on all process openings that are open or have temporary (tape and plastic) seals.

- Cut and seal all ductwork and conduit that penetrate the exterior concrete shell.

- Remove insulation from dome, sandblast and paint the surface.

- Install moisture detectors at the lower level of the building and in the sump to detect water inleakage. Install moisture detection alarms at the patrol station in Building 704-U.

- Shut off, close, and lock all building services (except for the moisture alarm system) such as steam, water, and electricity to prevent fire or water leakage in the building.

- Provide multiple locked barriers to prevent unauthorized entry of the building.

- Establish wells (assume three) to monitor ground water around the building.

Replacement of tape-and-plastic seals on process openings with permanent seals is intended to prevent access of water to the reactor. Removal of equipment and pipe sampling have caused obvious openings. However, other unknown openings probably exist, such as open sample valves, instrument tubing openings, or failed steam generator tubes. Some allowance is included in the cost estimate to develop a more comprehensive final design for water exclusion if this option is selected.

Sandblasting and painting the containment dome exterior surface after removing the adhesive bonded insulation are included to prevent further corrosion of the 3/4-inch steel shell (Figure 18). Moisture penetration of the insulation has caused significant pitting corrosion; up to 0.045-inch penetration was observed in 1964.

This alternative would require periodic inspection of the security fence, locked barriers, and the dome for corrosion. Some effort would be required to maintain the dome and the moisture detector system. Ready access to enter the building and investigate a moisture accumulation would be maintained.

Several options considered in developing this alternative, but not costed, are included here as additional information so that they can be reconsidered if protective confinement is selected.

- Pipes leading to the reactor could be cut off and sealed close to the reactor. For example, the 10-inch main system pipes could be cut off just outside the biological shield as specified for the entombment alternative. This option might be less costly and more effective than sealing the many process penetrations.

- With no insulation on the exterior of the dome the interior could reach high temperatures (perhaps 150°F) on hot clear days, and this could accelerate deterioration of the building, the equipment inside, and the building seals. Some options are:

  - Replace and maintain the dome insulation with new insulation that will not cause corrosion of the steel. Insulation might be placed on the inside surface.

  - Leave the existing insulation in place if the corrosion penetration rate is sufficiently low.

  - Provide vents for air circulation to prevent excess temperatures with no insulation.

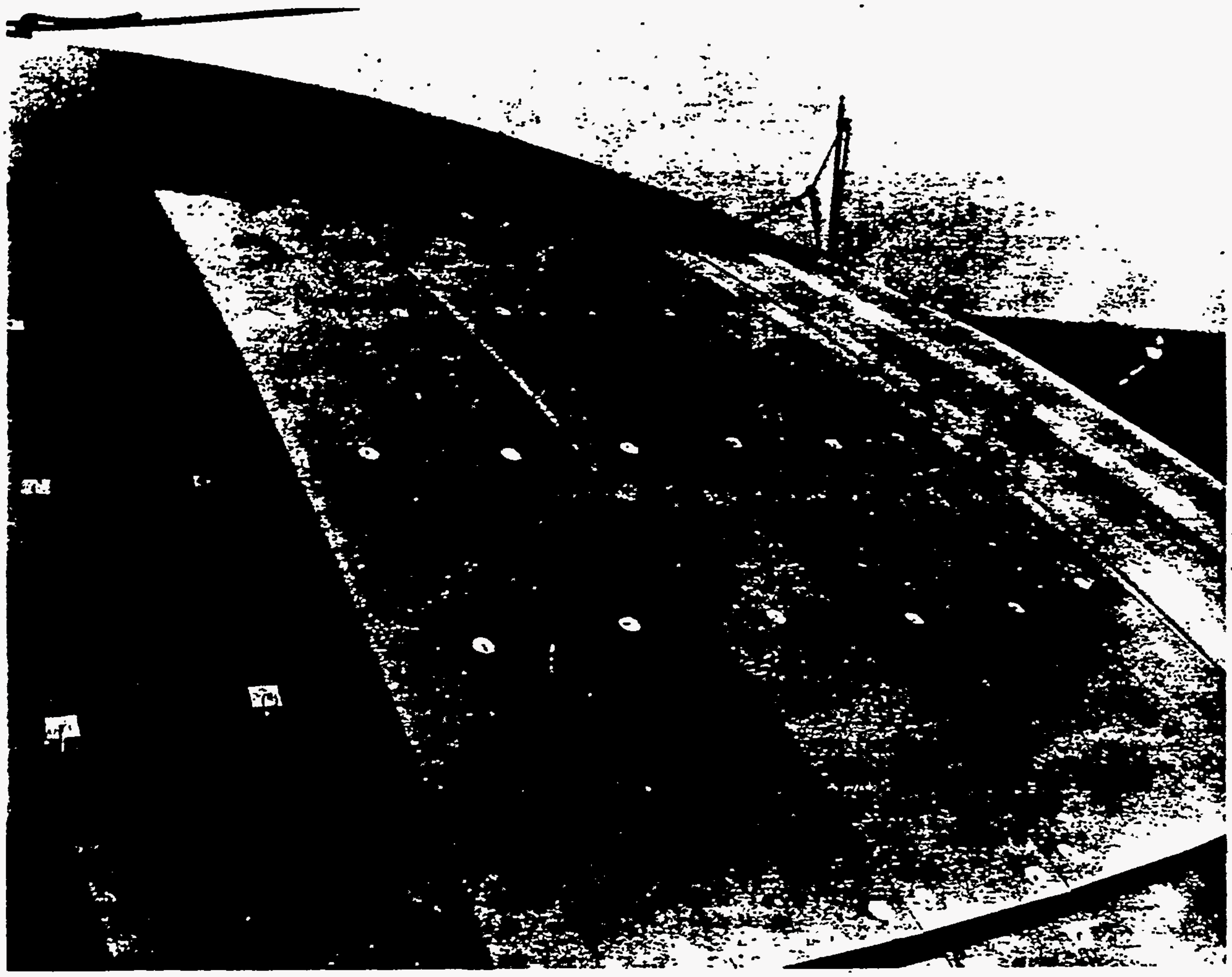


FIGURE 18. Insulation Being Installed on Containment Shell

SAVANNAH RIVER PLANT SITE

The Savannah River Plant site is shown in Figure 19. The site of HWCTR is at U area, over 3 miles from the site perimeter. Public entrances to the Plant are manned around the clock by security forces; public traffic traverses the Plant via route SC 125 that passes with 1-1/2 miles of HWCTR. Motorists are not to leave the public route; on-duty security personnel at the U site provide additional security for wayward sightseers. The HWCTR site is surrounded by a fence with a locked gate. The nearest stream to HWCTR is Upper Three Runs Creek.

The variety of nuclear operations have resulted in several activity sites as listed in Table 15. One location of special interest to the decommissioning plan is the burial ground.

The SRP burial ground occupies 195 acres located on high ground between the 200-F and 200-H Chemical Separations Areas. About 90 acres have been used to date, and the remainder is designed to provide space for disposal of contaminated wastes for an additional 22 years of SRP operation. The depth to the water table at the site is 40 ft; procedures require that no materials be buried closer than 10 ft from ground water. The surface water or ground water flows either to Four Mile Creek or to Upper Three Runs Creek depending on the burial site. Normally, materials are buried in trenches 20 ft deep by 20 ft wide, with at least 4 ft of soil cover. Segregation of wastes according to the type and extent of radioactive contamination is shown in Table 16.

Contaminated solid wastes from the HWCTR decommissioning would be stored at this location. Dismantlement would generate the most wastes (about 14,000 ft^3 containing about 2×10^4 Ci). These wastes are estimated to occupy about 2 acres or 1% of the space available at the site. About 3000 ft^3 would probably be classified as high-level waste; the remainder would be placed in low-level trenches. The high-level waste would include primarily the reactor vessel and materials from the primary coolant system. Through 1974, about 7×10^6 ft^3 of solid wastes contaminated with fission products and activation products have been buried at the storage site. As shown in Table 15, these wastes contain an estimated 10^6 curies of ^{60}Co. The additional quantities of wastes from HWCTR would be a small percentage of these values.

Commercial facilities for burial of contaminated wastes are also capable of accommodating wastes from dismantlement of a reactor such as the HWCTR. For example, the Chem-Nuclear Corporation's facility in Barnwell County, S. C., contains 270 acres and is designed for operation through 1993 with a capacity of 8.8×10^7 ft^3. Through 1973, over 800,000 ft^3 of waste has

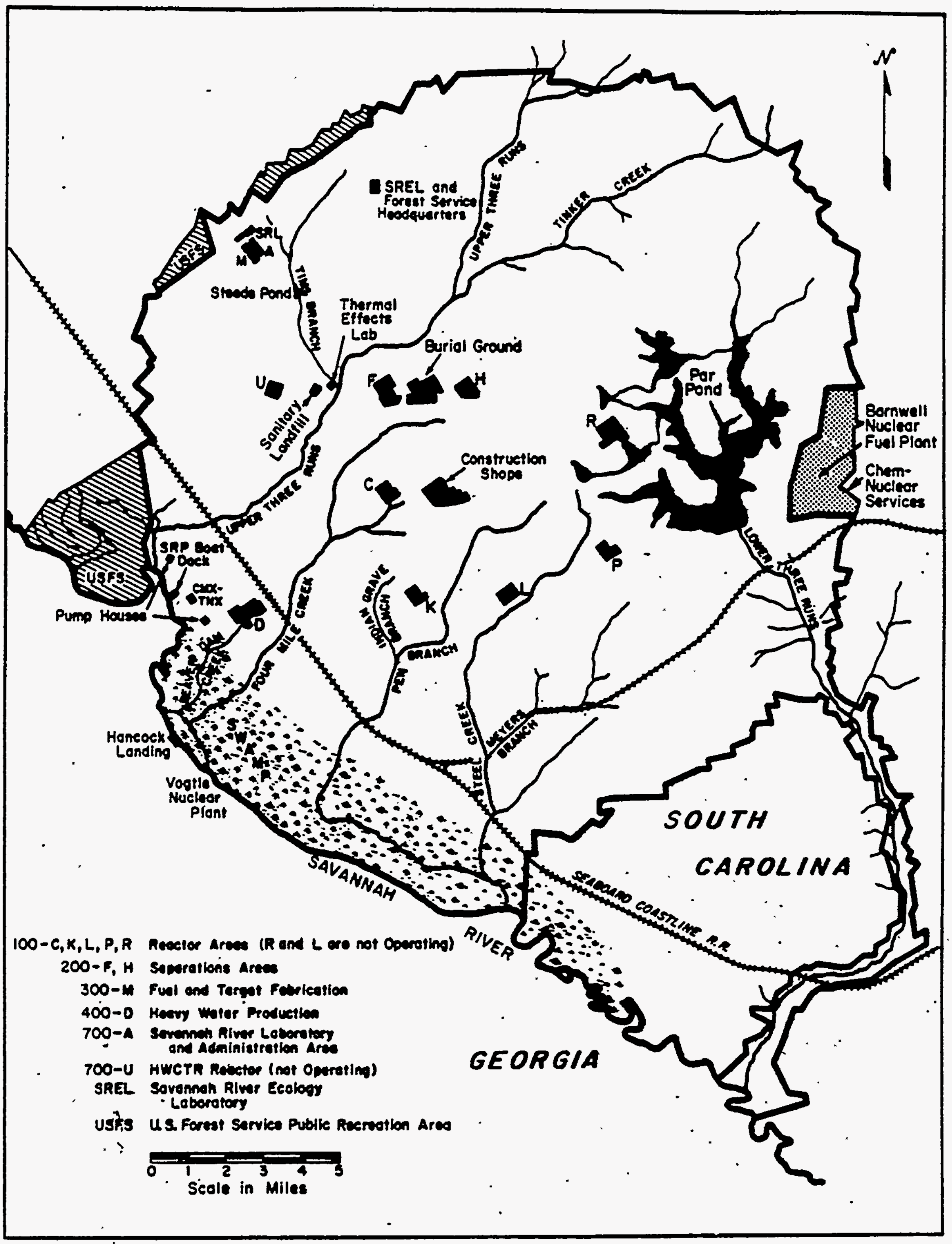


FIGURE 19. Savannah River Plant Site

been buried at this site, containing about 120,000 Ci of "byproduct" material (fission and activation products). Transfer of the reactor vessel to this site would require special approval by State authorities.

TABLE 15

Activity Sites at the Savannah River Plant[a]

		Activity, Ci (12/73)			
Location	*Size, acres*	*Fission Product*	^{60}Co	^{63}Ni	*TRU*
Burial Ground, F & H	90 used, 105 new	10^4	10^6	10^3	10^5
HWCTR	1	-	10^4	2×10^3	-

a. Other major activity sites include five reactor areas, two separations areas, and SRL; see Figure 19 for dispersed locations on plant site.

TABLE 16

Savannah River Plant Radioactive Solid Waste Disposal

Category	*Disposition*
Beta-Gamma	
Low Level Waste	
<50 mrem/hr at 3 inches <300 mrad/hr at 3 inches	Low-level trench
High Level Waste	
>50 mrem/hr at 3 inches >300 mrad/hr at 3 inches	High level trench
General Alpha Waste	Alpha trench
<10 nCi/g	
Retrievable Alpha Waste	55-gallon drums for pad storage
10 Mi/g to 0.1 Ci/pkg >0.1 Ci/pkg	55-gallon drum for storage in concrete container
Special	
Process equipment, vessels, jumpers, etc.	Plans developed on each case separately

SAFETY ANALYSIS

After decommissioning HWCTR, the residual radioactivity might cause a radiation dose to the public via two general mechanisms: accidental direct exposure to curious individuals who gain unauthorized access to the vicinity of the reactor vessel or accidental release of water that has contacted the reactor internals and then is released to the plant system. The likelihood of either occurrence is extremely small for all alternatives. Physical security associated with current site activity provides a high degree of protection against unauthorized access. No significant amounts of activity reach the plant streams under pessimistic consequence analyses that assume structural failures without corrective action. The safety analysis is summarized in Table 17.

TABLE 17

Decommissioning Safety Summary

Alternative	*Protection Against Direct Dose to Public*	*Offsite Water Transport to Public*		*Associated Retention Water Rights, acres*[a]
		Most Severe Consequence	*Probability*	
A-1 Dismantlement A-2	Physical security of SRP and burial ground	10^{-3} of guideline concentration	Very low - water table is 10-20 ft below reactor	Burial ground to creek (zero incremental to burial ground requirement)
B-1 Basic Entombment	Good - Not very dependent on physical security	10^{-5} of guideline	Low water table and very sound structure	From 0 to 90 acres depending on conservatism
B-2 Solid Entombment	Best - Independent of physical security	10^{-5} of guideline	Least probable	
C Protective Confinement	Physical security of SRP and HWCTR site	10^{-3} of guideline	Very low - water table is 50 ft below reactor	90 acres - HWCTR site to Upper Three Runs

a. Only meaningful if some reduction is made in the current SRP site.

Direct Exposure

The likelihood of radiation emitted from HWCTR causing any significant dose to the public after decommissioning is judged to be extremely low for any alternative.

Dismantling (A-1 or A-2)

If HWCTR were dismantled, the reactor vessel would be buried in a vault under 4 ft of earth in the site burial ground. Site security has been previously discussed. Admission to this SRP site is made only to security-cleared personnel or authorized visitors escorted by cleared personnel. Some closely regulated travel on a public road (SC 125) is supervised by SRP Security. Unauthorized entry to the burial ground would require breaching two fences; furthermore, direct exposure would require digging up the reactor vessel vault. The vessel vault could be filled with concrete to immobilize the activity and would also reduce the emitted radiation.

Entombment (B-1 or B-2)

Entombment is less reliant on physical site security to avoid dose-to-man because the activity is fixed in a concrete structure and the external dose is essentially background. The basic entombment structure at the HWCTR site would be inside the site security fence and would be much more formidable for entry. Ground level radiation would be background. Entry could be gained only by penetrating 1-3 ft of reinforced concrete. Solid entombment further reduces the likelihood of public access to the activity.

Protective Confinement

The protective confinement mode is most dependent on physical security to prevent exposure of an unauthorized member of the public. Multiple locked barriers are now and will continue to be provided. Presently the HWCTR site is surrounded by a locked security fence. Access to authorized personnel is permitted either by key or by security personnel through the office wing (Figure 1). The HWCTR site is further isolated from the public because of its location in the security area of the SRP site. If warranted under the protective confinement mode, the building doors and hatches could be welded shut or heavily locked. However, welded closures would interfere with surveillance of the system. Plant security personnel make frequent patrols inside the SRP site area.

For accidental or inadvertent radiation exposure to occur, some individual would have to break through a manned security fence, through a second locked fence or security checkpoint then deliberately break locks or welds to gain entrance to the HWCTR building. Inside the building the radiation levels in most areas are much less than 3 mR/hr. Any exposure to an unauthorized individual could be only from an incredibly determined and deliberate effort. Over 99.99% of the residual activity is contained in the reactor vessel, which is inaccessible without use of highly sophisticated tools and a heavy crane (now disabled) to lift the reactor head. The highest current level of radiation accessible in the building outside the biological shield is 200 mR/hr at the cyclone room sample lines. In 30 years, the radiation level will be <4 mR/hr (1 mR/hr after 40 years).

The physical security for all residual radioactive material from the HWCTR will exceed that prescribed in Regulatory Guide 1.86, *Termination of Operating Licenses for Nuclear Reactors*, under the category of "*possession-only license*." The Guide requirements state:

"a. Physical security to prevent inadvertent exposure of personnel should be provided by multiple locked barriers. The presence of these barriers should make it extremely difficult for an unauthorized person to gain access to areas where radiation or contamination levels exceed those specified for a dismantled facility (Table 8); these levels specified are about twice background. To prevent inadvertent exposure, radiation areas above 5 mR/hr, such as near the activated primary system of a power plant, should be appropriately marked and should not be accessible except by cutting of welded closures or the disassembly and removal of substantial structures and/or shielding material. Means such as a remote-readout intrusion alarm system should be provided to indicate to designated personnel when a physical barrier is penetrated. Security personnel that provide access control to the facility may be used instead of the physical barriers and the intrusion alarm system.

b. The physical barriers to unauthorized entrance into the facility, e.g., fences, buildings, welded doors, and access openings, should be inspected at least quarterly to assure that these barriers have not deteriorated and that locks and locking apparatus are intact."

Consequence Analysis

The activity remaining in HWCTR consists of ^{55}Fe, ^{60}Co, ^{63}Ni, and small amounts of plutonium from fuel failures. After briefly discussing the reasons why plutonium and ^{60}Co are not considered relevant to this analysis of offsite dose effects, the effects of highly unlikely failures postulated after decommissioning HWCTR are analyzed with emphasis on ^{63}Ni. Conservatisms used in the analysis are given in Table 18, and the results are compared in Table 19. The conclusions are:

- All alternatives involve a low risk of public exposure.
- Dismantlement centralizes surveillance at the site burial ground.
- Basic entombment fixes the activity but surveillance is limited; solid entombment provides the best security of HWCTR activity. The justification to retain water rights for entombment is questionable.
- Protective confinement is most dependent on current site size and vitality.

TABLE 18

CONSERVATISMS

- Very early failure of structure (20 years) with no corrective action.
- Corrosion rates of carbon or stainless steel increased by a factor of 10 over nominal values for cold water corrosion.
- All corrosion product dissolves or is suspended; no credit for large pieces of metal not transported by the water.
- No credit for ion exchange or retention of solid corrosion product (containing activity) by soil outside the decommissioning structure.
- Normal nickel concentration assumed for plant streams; Upper Three Runs should contain higher natural nickel from 300 Area operations, and thus uptake of radioactive nickel may be retarded.
- Water seepage from the structure is assumed to be replenished so that the maximum concentration of iron corrosion product in the water is maintained.

TABLE 19

Consequence Analysis

Alternative	*Maximum ^{63}Ni Concentration in Reactor Vessel, µCi/cc*	*Volume of Water Seeping to Creek, ft^3/yr*	*Maximum Creek Concentration, µCi/cc*	*Fraction of Guideline*
Dismantlement	5×10^{-5}	2,000	8×10^{-11}[a]	1/1500
Entombment	5×10^{-5}	1,300	5×10^{-12}	1/2500
Protective Confinement	1.8×10^{-4}	20,000	3×10^{-10}	1/400

a. Based on Four Mile Creek flow of 20 cfs; other concentrations are based on Upper Three Runs Creek flow of 200 cfs.

Plutonium (TRU)

The consequence of releasing the residual amounts of ^{239}Pu is negligible. Even if all the ^{239}Pu were released at one time, its effect would not be significant.

The residual plutonium in the HWCTR site is in corrosion products in the external piping from fuel failures (8-mg release, estimate). The exact amount remaining in the piping is difficult to estimate but is probably less than 10 mg of ^{239}Pu or 620 µCi (samples indicated 13 mg, 800 µCi).

If 10 mg of ^{239}Pu were dissolved in 3,700,000 gallons of water, the concentration would be about 4.4×10^{-8} µCi/cc, the guideline concentration. This volume of water corresponds to 40 minutes of flow in Upper Three Runs Creek or about five times the volume of the HWCTR building.

Therefore, ^{239}Pu does not contribute a significant amount of radioactivity to the postulated release model compared to ^{63}Ni or ^{60}Co.

^{60}Co

The radiation level in HWCTR is almost exclusively from ^{60}Co; thus, as ^{60}Co decays, the radiation levels will decrease. In 35 years, the radiation levels will be 1% of current values, and any construction work will be simplified greatly.

The consequence of ingesting water with ^{60}Co from water that contacts and corrodes the thermal shield of the HWCTR is currently about 40% of the total radiation burden to man. However, within 20 years, the contribution of the ^{60}Co becomes less than 5% of the total (and less than 1% in 35 years). None of the decommissioning alternatives involve credible release mechanisms on such a short time interval. Therefore, only ^{63}Ni will be considered in consequence analyses given in the next section.

^{63}Ni

The sequence of events discussed below shows that even for highly unlikely postulated failures the consequences to the public for release of radioactivity are significantly less than the guideline for any of the decommissioning modes (Table 19).

Dismantlement Release Model

1978

Reactor dismantlement complete. Pressure vessel removed intact and placed in vault in burial ground. Plugs in pressure vessel assumed to be at least equivalent to 0.2 inch of stainless steel in thickness. Immediate water invasion of vault is postulated. Vault is assumed to be 10 x 10 x 37 ft long.

Water starts to corrode through pressure vessel at thinnest section of stainless steel (assumes only 0.2 inch of stainless steel drainline). Corrosion rate is assumed to be 10 times the rate of general corrosion of stainless steel (10 x 0.0001 inch/year = 1 mil/yr). No credit is taken for saturation of water in vault with steel corrosion products. Chemical saturation of the water would arrest any further corrosion after several days when equilibrium is reached. Thus, with stagnant water in the vault, there is no mechanism for corroding through the pressure vessel. Further intrusion of fresh ground water is assumed (2000 ft^3/yr).

2178

Water penetrates into reactor vessel. The activity of the water inside the pressure vessel now increases to a maximum from corrosion of the stainless steel thermal shield. Again, no credit is taken for chemical saturation of this stagnant water which would inhibit further corrosion. Activity is 5×10^{-5} μCi/cc of ^{63}Ni in water in the vault. This activity is about 400 times the guideline concentration. The yearly rainfall associated with the vault surface area (400 ft^2) is about 2000 ft^3 or 15,000 gallons.

This water is postulated to seep out of the pressure vessel, through the reactor building and into the water table (10-20 ft below the vault).

No credit is taken for the vault that might provide additional holdup for the water and permit further radioactive decay. No credit is taken for slow transit time of the contaminated water from the vault to the water table.

2278

Water migrates to Upper Three Runs Creek. The migration time through the water table is calculated to be more than 100 years (115 to 150 years).

The activity of the ^{63}Ni in the creek is calculated assuming this 100-year migration (one half-life of decay) plus the dilution by the creek (factor of 3×10^6) assuming that the contaminated water enters the creek over a one-year period. No credit is taken for ion exchange in the soil between the reactor and the creek. Decontamination factors for soil of about 10^5 in 5 meters are reported in the literature. Such decontamination factors would result in virtually no measurable activity in the ground water just a few dozen meters from the burial ground site.

Activity in Upper Three Runs Creek is 8×10^{-12} µCi/cc of ^{63}Ni. This activity is about 1/15,000th of the guideline concentration. Therefore this release to the public would be negligible.

If the activity migrates to Four Mile Creek instead of Upper Three Runs Creek (depending on location in the burial ground), the activity concentration in Four Mile Creek will be about 10 times greater because of lower flow rate. This activity would be about 1/1500th of the guideline concentration and thus negligible.

Entombment

1978

Reactor entombed. Reactor vessel sealed with plugs welded closures at least equivalent to 0.2 inch of stainless steel in thickness. Reactor building not filled with concrete. This model analyzes basic entombment. It is therefore conservative for the solid entombment which is a more secure and substantial structure.

2000

Reactor building around pressure vessel fills with water and pressure vessel begins to corrode. This is a conservative assumption. No credible mechanism can be postulated for total roof failure this quickly. Entombment structures of this type have been designed for 100 to 140 year lives. Assuming (1) the roof fails, (2) water accumulates in the reactor building from rainfall of 60 inches/year (SRP average is less than 50 inches/year), (3) all rainfall over area of building going into building, (4) no water seeping out, (5) no water evaporating, the time required to fill the building to the level of the thermal shield would be about 5-10 years. With the purge rainfall assumed later, 100 yr would be needed and activity concentrations would be halved because of ^{63}Ni decay.

2200

Water in building corrodes through pressure vessel at thinnest section of stainless steel (assumes only 0.2 inch of stainless steel drain line). Assumed at 10 times the rate of general corrosion of stainless steel (10 x 0.0001 inch/year = 1 mil/yr). No credit is taken for saturation of water in building with steel corrosion products. This chemical saturation of the water would arrest any further corrosion after several days when equilibrium is reached. Thus without inflow of rainwater, there is no mechanism for the stagnant water in the reactor building ever corroding through the pressure vessel.

The activity of the water inside the pressure vessel now increases to a maximum from corrosion of the stainless steel thermal shield. Again, no credit is taken for chemical saturation of stagnant water that would inhibit any further corrosion.

Activity is 5×10^{-5} µCi/cc of ^{63}Ni in water in HWCTR building. This activity is about 400 times the guideline concentration (Table 9).

A water volume of 10,000 gallons per year is assumed to enter the reactor vessel.* A similar volume of dispersed water is postulated to seep out of the pressure vessel, through the reactor building and into the water table, which is 30 ft below the building foundation. No credit is taken for dilution of 4000 gallons of water in the pressure vessel (needed to submerge the highly activated shield in the lower half of the vessel) by the larger volume of uncontaminated water in the reactor building (about 10^6 gallons). This dilution would lower the concentration to the order of 10^{-8} µCi/cc which is less than the guideline concentration (Table 9).

No credit is taken for the concrete foundation which is substantial and would provide additional holdup time for the water and further radioactive decay. No credit is taken for slow transit time of the contaminated water from the building foundation to the water table or for unequal mixing.

---

* Purge volume corresponds to annual rainfall over a 9-ft radius circle (about to the area of the tomb roof structure shown in Figure 14)

2300

Water migrates to Upper Three Runs Creek. The migration time through the water table is calculated to be more than 100 years (115 to 150 yr). The activity of the ^{63}Ni in the creek is calculated assuming this 100-yr migration (half-life of decay) plus the dilution by the creek (5×10^6) assuming that the contaminated water enters the creek over a one-year period. No credit is taken for ion exchange in the soil between the reactor and the creek. Decontamination factors for soil in the order of 10^5 in 5 meters have been found in the literature. Such contamination factors would result in virtually no measurable activity in the ground water just a few dozen meters from the HWCTR site. Decontamination factors for the soil around the HWCTR have not been measured but could be considered for future plans if entombment or protective confinement is selected for decomissioning.

Activity in Upper Three Runs Creek is 5×10^{-12} µCi/cc of ^{63}Ni. This activity is about 1/2500th of the guideline concentration (Table 9); therefore this release to the public would be negligible.

Protective Confinement

Year

1978

Protective confinement completed. Reactor sealed by closing penetrations in carbon steel piping.

2000

Reactor building fills with water by dome failure plus no corrective action.

2006

Water in reactor building corrodes into pressure vessel. Corrosion path is shorter than entombment mode because it is through the 10-inch carbon steel line. The rate assumed is 10 times the general corrosion rate for carbon steel (10 x 0.006 inch/year = 60 mils/yr). As in the entombment case this takes no credit for corrosion decrease when the stagnant water becomes saturated by corrosion products.

Activity is 1.8×10^{-4} µCi/cc of ^{63}Ni in water (in reactor). This activity is about 700 times the guideline concentration (Table 9).

The building must fill with water before water can corrode into the vessel; rainfall (estimated as 20,000 ft^3/yr based on a 70-ft diameter circle, and 5 ft of rain) displaces this volume of water to the ground water annually.

As in the previous case of entombment, this water is postulated to seep out of the pressure vessel, through the reactor building, and into the water table which is 30 ft below the building foundation. No credit is taken for dilution of the water in the pressure vessel by the larger volume of water in the reactor building (about 10^6 gallons). Dilution would lower the concentration to the order of 10^{-7} μCi/cc which is about the guideline concentration; it is highly unlikely that the saturation value would be reached in all water in the building.

No credit is taken for the concrete foundation, which is a substantial structure that would provide additional holdup time for the water and thereby permit further radioactive decay. No credit is taken for slow transit time of the contaminated water from the building foundation to the water table.

2106

Water migrates to Upper Three Runs Creek. The migration time through the water table is calculated to be more than 100 yr (115 to 150 yr). The activity of the ^{63}Ni in the creek is calculated assuming this 100-yr migration (which is another half-life of decay) plus the dilution by the creek (3×10^5) assuming that the contaminated water enters the creek over a one-year period. No credit is taken for ion exchange in the soil between the reactor and the creek.

Activity in Upper Three Runs Creek is 3×10^{-10} μCi/cc of ^{63}Ni. This activity is about 1/400th of the guideline concentration; therefore this release to the public would be negligible.

The most severe effect postulated via natural events (earthquake, tornado, and flood) at the Savannah River Plant site is to initiate the failures in the decommissioned HWCTR structures that lead to the sequence of events described in the previous section. With onsite surveillance, the structural failures would presumably be repaired and the likelihood of activity transport eliminated.

Earthquakes

The Savannah River Plant is located in an area where moderate damage might occur from earthquakes based on earthquake risk predictions by the U. S. Coast and Geodetic Survey. On the basis of three centuries of recorded history of earthquakes, an earthquake above the intensity of Modified Mercalli (MM) VII would not be expected at the Savannah River Plant; the decommissioned HWCTR structures may not be damaged at all by this intensity of earthquake.

The MM VII earthquake is defined as an earthquake whose effect is:

> "Difficult to stand. Noticed by drivers of motor cars. Hanging objects quiver. Furniture broken. Damage of masonry of weak materials (such as adobe) or of poor mortar; masonry characterized by low standards of workmanship and weak horizontally. Some cracks in masonry of ordinary workmanship and mortar which is characterized as having extreme weaknesses such as failing to tie in at corners, not reinforced nor designed against horizontal forces."

Dismantlement

A reinforced concrete vault containing the reactor vessel in the burial ground would probably be undamaged by an earthquake. However, if required, the vault could be strengthened at nominal cost by pouring solid concrete around the pressure vessel for earthquake resistance and added resistance to water penetration.

Entombment

The concrete structure for basic entombment design (Figure 14), is probably adequate to resist a MM VII earthquake. If the HWCTR building were filled with concrete, no earthquake damage would be expected.

Protective Confinement

If confinement integrity is damaged by an earthquake, release of radioactive materials would be no more severe than that shown in the release model. Such damage would only be a method for the postulated structural failures in the model. Because site surveillance and structural maintenance are associated with protective confinement, earthquake damage could be repaired before activity transport.

Tornadoes

The residual activity at the HWCTR site is primarily induced in large structural steel members integral with a 100-ton reactor vessel. There is no credible mechanism to transport activity in this form via tornado winds. However, tornado-generated missiles can be postulated to cause limited structural failures to decommissioned structures. With site surveillance and maintenance, such damage could be repaired before any activity transport via water penetration of the breach occurs. Without corrective action, the consequence sequence previously described might be initiated.

The probability of a tornado as damaging as that used in the analysis is vanishingly small. The analysis is based on tornado wind velocities of 360 mph as postulated in Regulatory Guide 1.76. The probability of a tornado with wind velocities of 260 mph is only 1.0×10^{-4} per year. This is based on 22 years of tornado statistics for South Carolina and Georgia.*

Dismantlement

The components removed in dismantlement will be buried under several feet of earth in the burial ground. The pressure vessel will be further protected in a concrete vault. The combined protection of the pressure vessel by the vault and earth overburden would prevent tornado damage.

Entombment

Preliminary calculations show that the roof of the basic entombment structure will not be penetrated by postulated tornado missiles; the solid entombment is even more resistant to damage.

Protective Confinement

Preliminary calculations for protective confinement indicates that the dome may be penetrated by a tornado missile. Radioactivity would still be confined; however, the penetration would only allow rainwater penetration (a leaky roof). Because protective confinement is associated with surveillance and maintenance, this dome penetration (should it ever occur) could be repaired.

The conclusion, therefore, is that the risk to any mode of decommissioning from tornadoes is negligible.

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* D. W. Pepper, *Tornadoes: Characteristics, Probabilities, and Consequences of Occurrence at SRP*, DPST-74-563, December 20, 1974.

Floods

Flooding of the HWCTR site or the burial ground is not credible. The grade elevation of HWCTR is 280 ft MSL (feet above mean sea level). The burial ground grade elevation is at least 240 ft MSL. The maximum flood level of the Savannah River is calculated to be 168 ft MSL. Even if flooding did occur, the consequences would be less severe than the consequences from the radioactivity release model described. Such flooding would dilute the corrosion products and render them less hazardous than as described in the release model.

The maximum flood elevation, 168 ft MSL, is calculated by methods approved in Regulatory Guide 1.59, *Design Basis Floods for Nuclear Power Plants*. It is calculated from one-half the probable maximum precipitation peak discharge along with simultaneous failure of all upstream dams plus maximum wave runup from a 50 mph wind.

ASSESSMENT OF ALTERNATIVES

The three alternatives for decommissioning HWCTR were evaluated as summarized in Table 1, using the criteria previously developed. All are feasible and involve a very low risk of near-term public exposure. Land area commitments in all cases are small and are currently irrelevant considering the near-term future of the SRP site. Only protective confinement would seem to require retaining long-term water rights; most licensed power reactors have selected this approach although it may also be considered as an attractive interim option. The costs are highest for dismantlement, intermediate for entombment, and modest for protective confinement. Critics have claimed that decommissioning is as costly as building nuclear facilities; however, when inflation is allowed for in HWCTR, the highest cost decommissioning mode is only 20% of the construction cost ($9 million in FY 1960 escalated to $28 million in FY 1978), and entombment and protective confinement are proportionately lower. Flexibility ratings are lower for the entombment approach and higher for dismantlement (site reuse) or protective confinement (modify facilities or decommission differently). Aesthetics rate higher for dismantlement (and entombment) because of the appeal of grassland over man-made structures visible in protective confinement.

EPILOGUE

Some of the information required in making the HWCTR decommissioning study was difficult to obtain. Documenting information relevant to decommissioning when a site is operational would simplify the eventual decommissioning effort.

Several examples of information that would be particularly valuable are listed below:

- Obtain analyses of the precursors of neutron activation products in structural materials. Elements of interest include vanadium, chromium, manganese, iron, cobalt, and nickel. Any material that will receive significant neutron exposure should be analyzed, including concrete shielding.

- The neutron flux distribution in regions well outside the reactor core should be calculated and perhaps measured. A three-dimensional representation would be especially useful in calculating biological shield activity. Fluxes should be related to some absolute power and exposure.

- A detailed history should be kept of plutonium released from fuel failures with estimates of the removal efficiency of the purification system.

- A history should be kept of unusual events that may lead to the deposit of neutron activation products in unexpected places, e.g., the incident involving failure of the boiling loop bayonet.

Some examples of improved data on the mechanism of activity transport from decommissioned facilities are documented to aid other studies.

- Samples of metal removed from the hydraulic system of operating reactors could be analyzed for TRU, fission products, and corrosion deposits to avoid the need to cut samples from a shutdown reactor.

- Activity release from the metal samples to water could be determined in long-term corrosion tests to confirm the activity transport rate, to determine the self-limiting effects of metal solubility, and to determine the mobile (soluble and suspended) fractions of corrosion products.

- Means of further immobilizing the activity could be investigated.

APPENDIX A

RADIATION DATA

Radiation intensities from HWCTR in 1965 and 1969 from Health Physics survey records are compared with recent radiation measurements at the same location (Table A-1).

TABLE A-1

Radiation Level, mR/hr at 3 inches or c/m[a]

0'0" Elevation	*1/11/65*	*8/2/69*	*5/21/75*
Process Lines	6	2000 c/m	2000 c/m
Top of Spent Fuel Basin	5	6000 c/m	2000 c/m
Several Areas	1	-	500 c/m
Stored Equipment	-	10	10
EP 21.2	-	10,000 c/m	1000 c/m
Tank Top	-	3000-10,000 c/m	1000-2000 c/m
-16'3" Elevation	*1/22/65*	*8/22/69*	*5/21/75*
Right Pump Room			
EP 377.7 Main Loop	-	10	5
EP 21.2	-	15	5
Main System D_2O Line	-	10	5
BL Outlet Stub	1000	150	60
BL Inlet Stub	500	-	-
Process Lines Average	35	-	-
General Area	6-15	-	1-5
Left Pump Room			
Valve 1119	500	-	-
EP 105	60	5	3
Process Lines Average	35	-	-
General Area	3-25	-	1-5
EP 1061-1062	-	10	4000 c/m
EP 187	-	15	8
LL Line	-	20	12
-37'6" Elevation			
Right Generator Room			
EP 178.1, 178.2	-	4000 c/m	2000 c/m
Average Process Lines	35	10	5
General Area	2-6	-	1-2
Left Generator Room			
EP 54 and Adjacent Lines	800	120	50
EP 45 and Adjacent Lines	600	130	80
EP 20.1	-	10	5
EP 40.1 and 40.2	-	10	-
General Area	.5-25	-	2-5
-52'6" Elevation	*1/22/65*	*8/22/69*	*5/21/75*
Left Purification Room			
EP 194	500	15	10
EP 41	200	150	80
EP 53	40	20	5
General Area	5	-	1
Right Cyclone Room			
EP 101.1 and 101.2	10	-	2000 c/m
Sample lines	-	-	200
General Area	1	-	<1

a. All radiation intensities are in mR/hr at 3 inches from the equipment or area indicated unless otherwise identified. 4000 c/m =1 mR/hr.

Measurements made of the radiation emitted by the reactor vessel are listed in Tables A-2 and A-3; a short discussion of the measurements follows the data.

TABLE A-2

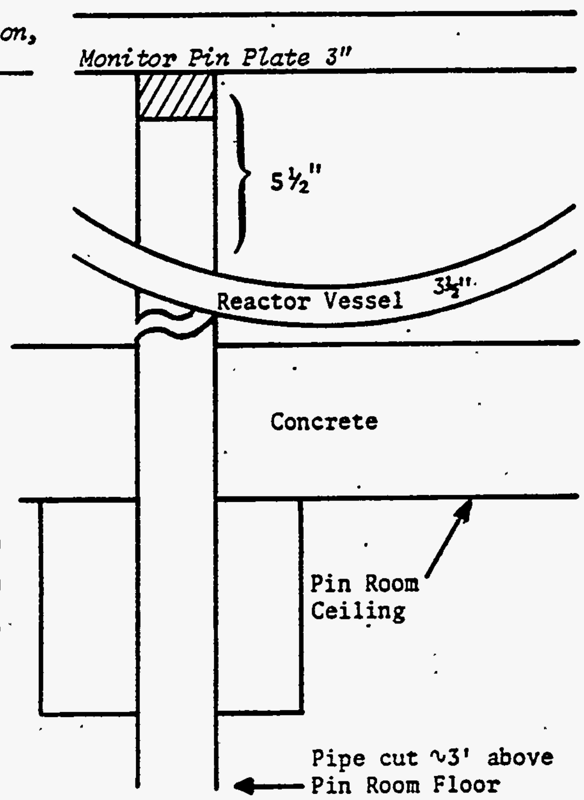
Radiation Measurements (with LND 716) in Power Level Sleeves

SLV-1, mR/hr	*Distance from SLV Opening, ft*	*SLV-4, mR/hr*
	11	<10
<10	12	15
35	13	35
100	14	130
300	15	210
500	16	210 (no window)
380 (carbon window)	17	150
85	18	85
20	19	35
10	19'6"	10

TABLE A-3

Radiation Measurements from Reactor Tank Bottom

Radiation, mR/hr	*Distance, ft*
140,000	
100,000	
7,000	1
1,600	2
300	3
210	4
210	5
180	6
130	7
85	8
	9
	10
	11
	12



Power Level Sleeve Measurements

In SLV-1, which has a 2 ft × 1.5 ft window box filled with carbon adjacent to the closest point of the sleeve to the reactor tank, the maximum radiation intensity was 500 mR/hr. At this point the detector was approximately 2 ft from the reactor tank. In SLV-4 at approximately the same location, the radiation intensity was 210 mR/hr. SLV-4 does not have the carbon-filled window box so there is approximately 1 ft of concrete and 1 ft of air between this sleeve and the reactor tank.

Reactor Tank Bottom

A gamma detector (LND-716) was inserted into a 1-inch pipe in the pin room. The pipe terminates inside the reactor vessel at the bottom of the monitor pin plate. The maximum radiation intensity detected was 140 R/hr inside the tank. At the outer surface of the tank about 7 R/hr was detected. This same pipe was used by Hochel for measurements with the Ge(Li) detector system.

A portable GeLi detector system was positioned at the end of the beam tube approximately 20 ft below the bottom of the reactor to identify the gamma emitters present. The only gamma emitter detectable was ^{60}Co and the activity was calculated to be 1.9×10^7 d/s/g assuming that the beam pipe end cap was 3 inches thick and that there was equal distribution of ^{60}Co.

Piping Samples

Data gathered from analyses of samples from the moderator systems of HWCTR are included in Tables A-4 and A-5. Estimates of total radioactivity in the various parts of the system are order of magnitude estimates.

TABLE A-4

HWCTR Moderator System Sample Data

Sample Locations	Sample	Portable Instrument Survey	Smear Inside Line		Analysis by Analytical Chemistry (PHA), d/m		
			βγ	α	TRU	^{137}Cs	^{60}Co
(10" line) Main System	3" metal plug*	30,000 c/m βγ	1100 c/m	300 d/m	3.3×10^3	$<9 \times 10^2$	9.9×10^6
(4"line) Liquid Loop	1.25" metal plug	4,000 c/m βγ	1300 c/m	-	6×10^3	$<3.7 \times 10^2$	1.3×10^6
(4" line) Boiling Loop	1.25" metal plug	15 mrad/hr at 3"	15 mrad/hr at 3"	-	10^4	$<9 \times 10^2$	8.4×10^6
Process Water Hold Tank (EP 41)	Two smears	25 mrad/hr at 3"	Same as sample		6.6×10^5	2×10^5	1.5×10^7
					2.8×10^5	6×10^5	3.6×10^7

* 3" diameter plug

TABLE A-5

HWCTR Moderator System Radioactivity

System	Area ft^2[b]	*Contamination/ft^2, d/m* TRU[c]	^{137}Cs	^{60}Co	*Estimated Total Contamination, Ci*[a] TRU	^{137}Cs	^{60}Co
Main System	5660	66×10^3	$<18 \times 10^2$	198×10^6	169.8×10^{-6}[d]	$<0.6 \times 10^{-6}$	0.5
Liquid Loop	700	720×10^3	$<44.4 \times 10^3$	166×10^6	229×10^{-6}	$<14 \times 10^{-6}$	.05
Boiling Loop	700	120×10^4	<10.9 10	1×10^9	381.8×10^{-6}	$<35 \times 10^{-6}$	0.32
Process Water Hold Tank EP	61.5[e]	9.4×10^5	4×10^5	2.5×10^7	26.3×10^{-6}	11×10^{-6}	696×10^{-6}
				Total	806.9×10^{-6}	11×10^{-6}	0.87

a. Based on the assumption that the contamination on the sample is representative of that in the system.

b. Appendix C

c. ∿90% ^{239}Pu and 10% ^{238}Pu

d. This is equivalent to 0.001×10^{-9} Ci/g of pipe

e. Area contaminated was assumed to be a strip covered with visible particulates ∿3' wide along the bottom of the tank.

Transferable Contamination

Health Physics completed a random disc smear survey of the zero level using about 150 disc smears. Smears were taken on the inside walls of the dome and equipment that could be reached from the floor or the first platform to the overhead crane. All of the smears were below the scaler background of 10 c/m $\beta\gamma$ and 3 c/m α with the exception of two smears in the vicinity of the tank top. These two smears were contaminated to 18 and 42 c/m $\beta\gamma$. No surveys were made of the dome or equipment at higher elevations because of poor lighting and means of access and may not be necessary until actual disassembly and removal of equipment begins. The fuel storage basin is covered with wood and a tarpaulin which were installed after reactor discharge. Health Physics survey records indicate that the basin was contaminated to 100 mrad/hr prior to the last flush before it was covered. Paper towel smears of the inside of the spent fuel basin indicated <100 α d/m/ft^2 and 200-400 $\beta\gamma$ c/m/ft^2 on the walls and 400-600 $\beta\gamma$ c/m/ft^2 on the floor areas. A portable instrument lowered to the -27' elevation of the basin floor indicated 15 mrad/hr at 2 inches.

Transferable contamination on external surfaces in process areas below the zero elevation is generally less than 500 $\beta\gamma$ c/m/ft^2 and less than 100 α d/m/ft^2. However, radiation survey records for scheduled work since the facility was shut down show that α contamination in the order of 40-50 α d/m/ 100 cm^2 to a maximum of 350 α d/m/ft^2 has been detected during line breaks, etc.

APPENDIX B

DERIVATION OF CONCENTRATION LIMITS

Dose Commitment

At SRP, the dose-to-man resulting from the release of radioactive species is calculated on the basis of the total dose over a 70-year period that results from a single year's release of the species. This concept is important when the combination of biological retention of the nuclide in the body and the nuclide's radioactive half-life result in a long-term effect from a short-term uptake. This is illustrated in the case of ^{63}Ni shown below.

^{63}Ni

Radioactive half-life	100 years
Biological half-life	800 days
Critical organ	bone
SRP Tech. Std. dose	30 mrem/yr
Limiting concentration in drinking water	5×10^{-7} µCi/cc

The cumulative dose for two hypothetical cases are given below. Case 1, one-year ingestion of the water is assumed; and Case 2, time is extended to 15 years.

Case 1

Drinking 1200 cc/day of water with $^{63}Ni = 5 \times 10^{-7}$ μCi/cc, for one year.

Year	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	Total
Bone dose, mrem	4.4	7.0	5.0	3.6	2.6	1.9	1.4	1.0	0.7	0.5	0.4	0.3	0.2	0.14	0.1	29.3

Case 2

Drinking 1200 cc/day of water with $^{63}Ni = 5 \times 10^{-7}$ μCi/cc, for 15 years.

Year	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	Total
Bone dose, mrem	4.4	11.4	16.4	20.0	22.6	24.5	25.9	26.9	27.6	28.1	28.5	28.8	29.0	29.1	29.2	352.4

In Case 1, the total dose commitment was 29.3 mrem, with a maximum annual dose of 7.0 mrem. In Case 2, the maximum was 29.2 mrem, with a total of 352.4 mrem. Continued intake beyond 15 years would result in a constant annual dose of 29.3 mrem. The Technical Standard requires that no annual exposure exceed 30 mrem; setting the limiting concentration on the basis of the dose commitment assures this (Case 2), but is conservative if releases are discontinued before the steady-state dose rate is reached (Case 1).

Concentration Limits

The 70-year dose commitment from ingestion of a radionuclide is given by

$$\text{Dose}_{70} = D_c \times C \qquad \text{(B-1)}$$

where

C = concentration in water, µCi/cc
D_c = dose conversion factor.

For the whole body and any organ except the GI tract,

$$D_c = \frac{kf\varepsilon}{m\lambda}\left(1 - e^{-2.555 \times 10^4 \lambda}\right) \qquad \text{(B-2)}$$

where

k = a constant related to rate of intake
f = fraction of radionuclide that reaches organ of interest
ε = effective energy in organ of interest, MeV
m = mass of organ of interest, g
λ = effective decay constant, days^{-1}

($\lambda = \lambda_r + \lambda_b$, where λ_r = radioactive decay constant and λ_b = biological decay constant) 2.555×10^4 = days in 70 years.

For the GI tract,

$$D_c = \frac{kf\varepsilon Te^{-\lambda_r t}}{m} \qquad \text{(B-3)}$$

where

k = a constant depending on mode of intake
T = residence time in portion of GI tract involved, days
t = time for ingested material to reach portion of GI tract considered, days

Other constants are defined for Equation B-2.

In calculating the dose from eating fish, the D_c value is multiplied by a concentration factor to convert µCi/cc of water to µCi/g of fish in equilibrium with water. The concentration factors vary depending on the element being considered.

Dose conversion factors for the nuclides of concern in the HWCTR were either obtained from Reference 1 or calculated using the parameters given below.

General Constants[1]

Vector	*Intake Rate/day*	*k (Eq. B-2)*	*k (Eq. B-3)*
Drinking Water, adult	1200 ml	2.22×10^{7}	1.1×10^{7}
Fish, adult	32.4 g	6.0×10^{5}	3.0×10

Organ	*Mass, g*
Whole Body	7.0×10
Bone	7.0×10
Large Lower Intestine (LLI)	150

GI Tract	*t, days*	*T, days*
LLI	0.542	0.75

Nuclide-Specific Constants[2]

Whole Body	^{60}Co	^{63}Ni	^{239}Pu
f	0.3	0.3	3.0×10^{-5}
ε	1.5	0.021	53
λ	7.3×10^{-2}	1.1×10^{-3}	1.1×10^{-5}
Bone			
f	-	0.15	2.5×10^{-5}
ε	-	0.11	270
λ	-	8.9×10^{-4}	9.6×10^{-6}
LLI			
f	1.0	-	-
ε	0.44	-	-
T_r	3.6×10^{-4}	-	-

Concentration Factors in Fish[3]

Element	*Factor*
Co	20
Ni	100
Pu	3.5

Dose Conversion Factors

Whole Body	^{60}Co	^{63}Ni	^{239}Pu
Drinking Water	2000	1800	11,000
Eating Fish	1060	4900	4,000
Bone			
Drinking Water	-	59,000	470,000[1]
Eating Fish	-	160,000	200,000
GI TRACT (LLI)			
Drinking Water	24,200	-	-
Eating Fish	13,200	-	-

The dose conversion factors were then used in Equation B-1, together with the limiting dose, to determine the appropriate concentration limits for this study. The whole body dose also contributes to the specific organ dose and must be included in determining total dose. Equation B-1 thus took the general form:

$$C_w = \frac{0.03}{D_{c_1} + D_{c_2} + D_{c_3} + D_{c_4}} \tag{B-4}$$

where

C_w = concentration in water, Ci/cc
D_{c1} = dose conversion factor, specific organ, for drinking water
D_{c2} = dose conversion factor, specific organ, for eating fish
D_{c3} = dose conversion factor, whole body, for drinking water
D_{c4} = dose conversion factor, whole body, for eating fish
0.03 = mrem limit for bone or LLI

For the case of drinking water only, D_{c2} and D_{c4} are deleted. The concentration limits calculated using Equation B-4 are given as follows:

Nuclide	C_w, µCi/ml Drinking Water Only	Drinking Water and Fish
^{60}Co	1.1×10^{-6}	7.4×10^{-7}
^{63}Mo	4.9×10^{-7}	1.3×10^{-7}
^{239}Pu	6.2×10^{-8}	4.4×10^{-8}

The resulting maximum annual doses at these concentrations are given as follows:

Drinking Water Only

Nuclide	mrem/yr Whole Body	Bone	LLI	Total to Organ
^{60}Co	2.2	-	26.6	28.8
^{63}Ni	0.9	28.9	-	29.8
^{239}Pu	0.7	29.3	-	30.0
Water and Fish				
^{60}Co	2.3	-	27.7	30.0
^{63}Ni	0.9	28.5	-	29.4
^{239}Pu	0.7	29.5	-	30.2

REFERENCES

1. "Appendix G - "Methods for Determining Environmental Radiation Dose," *Environmental Statement - Waste Management Operations*. Savannah River Plant, Aiken, South Carolina (Draft) USERDA Report ERDA-1537 (to be issued).

2. "ICRP Publication," *Health Physics 3*, June 1960.

3. S. E. Thompson, et al. *Concentration Factors of Chemical Elements in Edible Aquatic Organisms*, UCRL-50564, Rev. 1, October 1972.

APPENDIX C

MISCELLANEOUS EQUIPMENT DESCRIPTION

A. Reactor Vessel and Internal Parts (EP-1)

1. References

   BPF 210650
   Purchase order AX C24497½
   W230739
   PASECO supplied vessel and internal parts.

2. Weight

   The purchase order states that the calculated weight of the vessel and all internal parts is 98 tons (this excludes fuel but includes all other internals). Calculations made as a part of this study indicate the vessel weights 50 to 55 tons, and the internals weigh about 27 tons. The calculated weight of the internal parts is as follows:

		Pounds
Radial thermal shield plates		24,400
Bottom plate		1,600
Horizontal thermal shield segments		7,000
Top shield		5,600
Indexing shield plug		7,800
Shield plug support ring		8,000
	Total	54,400

B. Lower Axial Shield

1. References

   BPF 210650 (Part of EP-1)
   W231697
   D111481

2. Size

   58" OD × 37" high

3. Contents and Weight

The space among sleeves is filled with concrete. A memo from Kamack to Overbeck dated 9/28/61 apparently supersedes the drawings regarding concrete fill. The memo specifies the fill composition as follows:

		Pounds
Steel shot		8,507
Portland cement, Type 1		848
Water		331
	Total	9,747

The drawings specified only concrete as the fill. The empty weight of the shield is specified as 2500 lb on D111481; the total weight is about 12200 lb (D111481 also specifies a concrete fill volume of 42 ft^3, concrete density of 332 lb/ft^3, and a filled assembly weight of 4550. This weight is inconsistent with the volume and density; the volume appears to be correct and the weight wrong.)

4. Material

The shield is constructed of carbon steel. D111841 specified cadmium plating the sleeve inner surfaces and painting the exterior of the shell.

5. The shield is supported by 12 support lugs that form a segmented ring. The lugs are bolted to the pin room ceiling.

C. Steam Generators (EP 20.1 and 20.2)

1. References

BPF 120628

2. Description

• Empty weight	37,800 lb each
• Heat transfer surface area	2,500 ft^2 each
• Tubes	3/4" OD, 12 BWG, 0.109" wall ASTM SA-210 carbon steel
• Shell	1-1/8" thick wall } carbon steel 2-3/8" thick head } carbon steel
• Minimum carbon steel thickness to penetrate shell	10" Sch. 60 weld caps, 0.500" wall 2" drain pipe, 0.30" wall

D. Main System 10" Piping

1. Outside Biological Shield

- Sch. 100, 0.718" wall
- ASTM A106 Grade B carbon steel
- 220 ft purchased
- All connections are welded

2. Inside Biological Shield

- Sch. 100 304 stainless steel
- 24 ft purchased

APPENDIX D

PARTIAL LIST OF REFERENCE NUMBERS FOR HWCTR EQUIPMENT

Name	*Equipment Piece (EP) Number*	*Blue Print File (BPF) Number*
Reactor Vessel	1	210650
Steam Generators	20.1, 20.2	210628
Main Storage Tank	41	210588
Main Pumps	21.1, 21.2	210652
Gas Compressors	86.1, 86.2	210953 210954
ICL Storage Tank	194	
Makeup Pump	42	210610
SFB Deionizer	103	210757
SFB Filter	104	
LL Pumps	186.1, 186.2	211251
BL Pumps	178.1, 178.2	210651
Main System Deionizer	44.1, 44.2	210757
Main Purge Cooler	40.1, 40.2	
Seal Pot	43	210698
Hold Tank	105	210821
LL Cooler	187	
Hold Tank	53	210821
SFB Cooler	101.1, 101.2	210760
Drain Tank	51	
Purification Tank	47	210682
Catch Pot	92	210786
ICL Seal Pump	180, 195	
ICL Afterfilter	45	210740
LL Purge Cooler	191	
Transfer Coffin	270	
Rod Drive Platform	256	
Spent Fuel Basin Gantry	278	
Reactor Vessel	1	210650
Steam Generators	20.1, 20.2	210628

Name	*Equipment Piece (EP) Number*	*Blue Print File (BPF) Number*
Main Storage Tank	41	210588
Main Pumps	21.1, 21.2	210652
Gas Compressors	86.1, 86.2	210953, 210954
ICL Storage Tank	194	
Make-up Pump	42	210610
SFB Deionizer	103	210757
SFB Filter	104	
LL Pumps	186.1, 186.2	211251
BL Pumps	178.1, 178.2	210651
Main System Deionizer	44.1, 44.2	210757
Main Purge Cooler	40.1, 40.2	
Seal Pot	43	210698
Hold Tank	105	210821
LL Cooler	187	
Hold Tank	53	210821
SFB Cooler	101.1, 101.2	210760
Drain Tank	51	
Purification Tank	47	210682
Catch Pot	92	210786
ICL Seal Pump	180, 195	
Seal Head Tank	22	210694
Vent Condenser	84	210770
Separator	84.1	210770
ICL Seal Head Tank	181, 198	
Poison Tank	60	210696
Main Relief Valves	2, 57	210715
Rotary Bridge Crane	519	210626
Crane Railway		210633
10" Motor Operated Valves		210726
6" Motor Operated Valve		210838

Abbreviations

ICL	Isolated Coolant Loop
LL	Liquid Loop
BL	Boiling Loop
SFB	Spent Fuel Basin

APPENDIX E

EQUIPMENT REMOVED FROM HWCTR

EP No.	*Name*
84	Gas Compressor
48	Seal Supply Pump
190	Liquid Loop Surge Tank
175	Boiling Loop Surge Tank
177	Boiling Loop Cooler
169	Boiling Loop Purge Cooler

APPENDIX F

SURFACE AREAS OF PROCESS EQUIPMENT

1. Surface Exposed to D_2O (for alpha activity estimate)

   a. Carbon Steel Piping

      10" Main Loop Pipes, ~250 ft, 660 ft^2

      Pressure Relief and Vent Pipes
      12", ~33 ft, 104 ft^2
      8", ~60 ft, 126 ft^2

   b. Steam Generators (tube side), 2500 ft^2 each (carbon steel tubes)
      5000 ft^2 total

   c. Reactor Vessel, ~800 ft^2 (stainless steel)

   d. Reactor Internal Parts, ~1500 ft^2 (normally submerged)

   e. Purge Cooler (EP 40) ~600 ft^2 (carbon steel tubes)

   f. Liquid Loop
      Cooler (EP 187) 65 ft^2 } stainless steel
      Piping 500 ft^2 } stainless steel

   g. Boiling Loop
      Piping 500 ft^2} stainless steel

2. Stainless Steel Surface in Reactor Vessels (for ^{63}Ni release models)

   a. Inner surface of thermal shield plates having the maximum specific activity of ^{60}Co and ^{63}Ni (5 x avg):

	Square Feet
Side plates	160
Angle plates	37
Bottom plate	14
Top plate	14
	225

   b. Outer surface of thermal shield and reactor lining surface (in core region) having ^{60}Co and ^{63}Ni specific activity 100 times lower than inner surface of thermal shield plates:

      Area = 567 ft^2

c. Additional stainless steel surface area in reactor in upper region above the active core (assuming all surfaces submerged):

Area = ∿1500 ft^2

d. Weighted average ^{63}Ni and ^{60}Co specific activity, including nonactive stainless steel in reactor:

- Include all stainless steel in core region.

$$(SA)_{WA} = (SA)_{max} \frac{1 \times 225\ ft^2 + 0.01 \times 567\ ft^2}{225\ ft^2 + 567\ ft^2}$$

$$(SA)_{WA} = (SA)_{max} \quad (0.3)$$

where: $(SA)_{max}$ = maximum surface specific activity

$(SA)_{WA}$ = weighted average surface specific activity

- Include all stainless steel in reactor.

$$(SA)_{WA} = (SA)_{max} \frac{1 \times 225\ ft^2 + 0.01 \times 567\ ft^2}{225\ ft^2 + 567\ ft^2 + 1500\ ft^2}$$

$$(SA)_{WA} = (SA)_{max} \quad (0.10)$$

- Note that $(SA)_{max} = 5 \times (SA)_{avg}$

  where $(SA)_{avg}$ = average specific activity of stainless steel in core region

- The units of (SA) are μCi/g Fe.

e. The significance and application of the weighted average specific activity $(SA)_{WA}$ derived in item d above are as follows:

- Release of ^{60}Co or ^{63}Ni depends on corrosion of the stainless steel.

- The maximum ^{60}Co or ^{63}Ni content of the water in the reactor vessel is determined by the saturation concentration of Fe in the water and the weighted average specific activity of ^{60}Co or ^{63}Ni in the stainless steel; e.g.,

$$\mu Ci\ ^{63}Ni/cc\ H_2O = \frac{\mu Ci\ ^{63}Ni}{g\ Fe} \times \frac{g\ Fe}{cc\ H_2O}$$

  This assumes that all of the Ni and Co released by corrosion of stainless steel dissolve in the water.

- Exposing more nonactive stainless steel or Fe to the water lowers $(SA)_{WA}$ and lowers the content of ^{60}Co and ^{63}Ni in the water. The ratio $(SA)_{WA} = 0.3\ (SA)_{max}$ was used in the activity release calculations.

- One way to reduce the potential ^{60}Co or ^{63}Ni content of the water would be to fill the reactor with iron (e.g., steel pipes).

APPENDIX G

REFERENCES

W230739 - Cross Section of Reactor
W231080 - Reactor Piping Imbedded in Concrete, Plan
W231081 - Reactor Piping Imbedded in Concrete, Section
W231292 - Concrete around Reactor
W231697 - Lower Axial Shield
D111481 - Lower Axial Shield

W230829 Equip. Arrgt. at El. 0'-0"
W230849 - Equip. Arrgt. at El. -16'-3"
W230850 - Equip. Arrgt. at El. -37'-6"
W230851 - Equip. Arrgt. at El. -52'-6"
W230916 - Equip. Arrgt. at El. Sections
W230889 - Equip. Arrgt. at El. Sections
W230917 - Equip. Arrgt. at El. Sections
W230856 - Spent Fuel Basin
W168483 - Transfer Coffin

Map 3420, Sht. 1 - Plot Plan

Model - HWCTR Model is in the 773A Fab Lab

DPSTN 2535 contains

a. Historical memos/letters on HWCTR decommissioning

b. 1975 memos on HWCTR decommissioning

c. Miscellaneous HWCTR memos

d. Calculations

e. Notes

f. HWCTR photograph numbers

g. HWCTR drawing schedule