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ACTIVATED CORROSION PRODUCT
RADIATION LEVELS NEAR FFTF
REACTOR AND CLOSED LOOP
PRIMARY SYSTEM COMPONENTS

HANFORD ENGINEERING DEVELOPMENT LABORATORY
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1.0 SUMMARY

This report contains estimates of the radiation levels in an FFTF reactor Heat Transport System (HTS) cell and near a Closed Loop System (CLS) module due to activated stainless steel corrosion product deposits in primary system piping and components. ^{60}Co , ^{58}Co , ^{54}Mn , ^{51}Cr , ^{59}Fe , and ^{182}Ta were considered as corrosion products for this analysis.

In an HTS cell, the general radiation levels due to activated corrosion products were predicted to be in the range of 1-15 R/hr; near a CLS module the levels were predicted to be 30-1000 mR/hr, both after five years of reactor operation at ultimate conditions. The corrosion product radiation levels in an HTS cell were predicted to be high enough to interfere with contact maintenance. Levels near a CLS module may or may not affect contact maintenance.

Selective shielding of single, major components in either an HTS cell or a CLS module was shown to be ineffective because of the wide distribution of corrosion product deposits (and hence radiation sources).

Reducing the oxygen level of the sodium was predicted to cause a reduction in the general radiation levels. However, there is only meager experimental evidence to support this prediction and it must be used cautiously.

Lastly, this report deals only with radiation due to activated corrosion products. Radiation due to activated sodium or to fission products from failed or vented fuel was not considered.

2.0 INTRODUCTION

In the Liquid Metal Fast Breeder Reactor Program (LMFBR), the choice of material for fuel cladding, hardware and sodium containment is austenitic stainless steel, and for liquid metal coolant the choice is sodium. At the present level of technology these selections give the maximum combination of safety, reliability, and economics for the program. It is within the limits of present technology to purify sodium to the required levels for use as a coolant and to obtain stainless steel of the necessary properties. Under proper conditions the corrosion rate of the stainless steel in the sodium is low enough so that the structural integrity of fuel cladding, pipes, hardware, etc. can be maintained. However, the corrosion rate of the stainless steel is still not zero, and the small amount of stainless steel corrosion products released from the fuel cladding and other stainless steel surfaces present the problems which will be discussed in this report.

In the FFTF, the stainless steel corrosion products released from core surfaces (primarily fuel cladding) to the coolant will be highly radioactive. Radioisotopes such as ^{58}Co , ^{54}Mn , and ^{60}Co will be produced in the in-core steel by neutron activation of atoms in the stainless steel. Thus, even though the material removal rate from the core is so small that cladding and structural integrity will be maintained, the release and transport of activated corrosion products to all parts of the Primary Heat Transport System (HTS) will cause radiation fields near primary system components which could be high enough to affect maintenance.

To a lesser extent, a similar situation exists in the primary heat transport systems of the FFTF Closed Loops. Activated corrosion products from experimental fuel assemblies in the CLIRA (Closed Loop In-Reactor Assembly) will be transported throughout the closed loop heat transport system where they will produce radiation fields near closed loop components. Again, these radiation fields could be strong enough to hinder maintenance on closed loop components.

This report presents estimates of the radiation fields near reactor and closed loop primary heat transport system components which are caused by deposition of activated corrosion products released from in-core stainless steel surfaces. These estimates will be used by others to evaluate FFTF maintenance procedures and the requirements for temporary shielding of primary system components.

3.0 CALCULATIONAL MODEL

The mechanisms of release, transport, and deposition of activated corrosion products in the primary systems of sodium cooled reactors are complex phenomena not yet fully understood. Therefore, for the purposes of this study, a simplified model based on mass balances of corrosion products was used to describe these processes. A detailed description of the model is presented in Appendices A and B. The same basic model was used to describe the behavior of corrosion products in both the reactor and closed loop primary sodium systems. In the brief description that follows, the

terminology of the reactor primary heat transport system is used. However, a description of the model for the closed loop primary heat transport system involves little more than a change in nomenclature and system geometry.

Briefly, the model is logically divided into three parts: source, deposition, and dose rate calculations. The source section models the build-up of radionuclides in stainless steel due to the neutron activation of several of the constituents of the steel. For this analysis, ^{60}Co , ^{58}Co , ^{54}Mn , ^{59}Fe , ^{51}Cr , and ^{182}Ta were considered as corrosion products. The sources of these nuclides are given in Table 1.

Table 1
Corrosion Product Nuclide Sources

<u>Nuclide</u>	<u>Formation Reaction</u>	<u>Half Life</u>	<u>Source of Parent</u>
^{60}Co	$^{59}\text{Co}(n,\gamma)^{60}\text{Co}$	5.24 yr	Co impurity in steel.
^{60}Co	$^{60}\text{Ni}(n,p)^{60}\text{Co}$	5.24 yr	Ni in steel
^{58}Co	$^{58}\text{Ni}(n,p)^{58}\text{Co}$	71 d	Ni in steel
^{54}Mn	$^{54}\text{Fe}(n,p)^{54}\text{Mn}$	313 d	Fe in steel
^{59}Fe	$^{58}\text{Fe}(n,\gamma)^{59}\text{Fe}$	45 d	Fe in steel
^{51}Cr	$^{50}\text{Cr}(n,\gamma)^{51}\text{Cr}$	28 d	Cr in steel
^{182}Ta	$^{181}\text{Ta}(n,\gamma)^{182}\text{Ta}$	115 d	Ta impurity in steel

The (n,p) reactions occur in the fast flux of the core; the (n, γ) reactions occur both in the core and the regions adjacent the core. The source section of the model also covers the release of the corrosion product nuclides to the sodium coolant due to the corrosion of the activated stainless steel surfaces. The

release rate of each activated corrosion product nuclide is based on the average corrosion rate of the steel, with each nuclide being released in proportion to its fractional concentration in the steel. Adjustments to this "stoichiometric" release rate are required for some corrosion product nuclides where definitive data are available.

The deposition section of the model describes, as a function of time, the accumulation of activated corrosion products in the primary system which are released from the steel in the core region. At any given time, the total inventory of these corrosion products is assumed to be distributed throughout the primary heat transport system, deposited on all surfaces in the system. The distribution pattern of each corrosion product nuclide throughout the HTS is determined by consideration of the relative surface areas, temperatures, flow conditions, etc. of each major system component and by information from test loops and operating sodium-cooled reactors.

The final section of the model deals with the calculation of radiation fields due to the activated corrosion products deposited in major primary system components. Since these radiation fields are dependent upon the inventory and spatial distribution of the corrosion products throughout the primary system, a three-dimensional, geometrical model of a reactor HTS cell was developed to describe the spatial relationships of the corrosion product deposits within that cell. This model, together with inputs from the source and

deposition calculations is used to calculate radiation field intensities as a function of position within an HTS cell. Radiation dose rates are calculated at nine points in an HTS cell (six points near a CLS module). Details of the dose rate calculational model are given in Appendix B.

4.0 RESULTS AND DISCUSSION

The predicted radiation levels in and around an HTS cell and a CLS module are, of course, the primary "results" of this analysis. However, it is necessary to examine the assumptions and input information upon which the predictions are based in order to be able to effectively use those predictions.

4.1 Release of Corrosion Products to Coolant

The radiation levels predicted in this analysis are directly proportional to the inventories, and hence release rates, of each of the corrosion product radionuclides. (See model, Appendix A.) The release rates of these radionuclides are assumed to be proportional to the average corrosion rate of the steel, with each nuclide being released in proportion to its fractional concentration in the steel. This stoichiometric release rate must be adjusted for the minor constituents of the steel, cobalt, manganese, and tantalum, because the average corrosion rate is primarily a function of the release of iron, chromium and nickel, the major constituents

of stainless steel. These adjustments are based on observations from experimental test loops and operating sodium-cooled reactors. Those observations indicate that:

- Manganese is preferentially leached from stainless steel surfaces at about double the stoichiometric release rate.
- Manganese release rate is sensitive to the oxygen level of the sodium. A low oxygen level (1-2 ppm) is predicted to reduce the manganese release rate by about a factor of four relative to the release rate at a 5 ppm oxygen level.
- Cobalt is preferentially retained, i.e., cobalt corrodes at a slower rate than stainless steel. ^{58}Co was observed to be present in proportionally smaller amounts than ^{60}Co . The probable reason for this is that the nickel parent nuclide of ^{58}Co is preferentially leached from stainless steel by sodium, hence, the source is reduced. The cobalt release rate does not appear to be greatly affected by oxygen level.
- In spite of the apparently large potential for ^{182}Ta formation, this nuclide has not been found in EBR-II, and only in small quantities in LAMPRE which has Ta in the core. It is known that the corrosion rate of Ta in low oxygen sodium is quite low.
- ^{59}Fe and ^{51}Cr are assumed to be released at their stoichiometric rates. At low oxygen levels, the release rate of these nuclides (and the average corrosion rate) is predicted to decrease by a factor of four relative to the release rate at a 5 ppm oxygen level.

For the purposes of this study two cases were evaluated, one for oxygen levels of about 5 ppm and one for oxygen levels

below 2 ppm. These two cases were applied to both the reactor and closed loop primary systems. In all cases, except for the radial reflectors, the source material was assumed to be 316 stainless steel with the following nominal composition: 65% iron, 12% nickel, 18% chromium, 2% manganese, 2-3% molybdenum, 1% silicon, 0.02% cobalt, and 0.01% tantalum. (Although FFTF fuel cladding will be purchased to an 0.05% maximum cobalt specification, 316 stainless steel fuel cladding procured by HEDL to that specification usually contains about 0.02% cobalt.) The radial reflectors were assumed to be Inconel-600, with a low cobalt content (0.02%).

The preceding discussions apply generally to both the reactor and closed loop primary systems. Specific details of the release of corrosion products in these systems are presented below.

4.1.1 Reactor Primary Heat Transport System

Activated stainless steel corrosion products are released from the fuel cladding, wire wraps, fuel subassembly ducts, and the radial reflectors to the sodium coolant. Since the temperature of and sodium velocity over the steel surfaces, and nuclear parameters vary greatly throughout the core region, this source region was divided into four zones, and average corrosion rates, fluxes, and cross-sections were assigned to each zone. The assumed stainless steel surface areas and corrosion rates for each zone are presented in

Table 2. (For more details, see Appendix A.)

<u>Table 2</u>			
<u>HTS Source Zone Surface Areas and Corrosion Rates</u>			
<u>Zone</u>	<u>Surface Area</u>	<u>Avg. Corrosion Rate (mils/yr)</u>	
		<u>Case 1*</u>	<u>Case 2**</u>
Core	4500	0.055	0.014
Axial Reflectors	750	0.13	0.03
Gas Plenum	5250	0.095	0.024
Radial Reflectors	1500	0.025***	0.006

* Based on 1100°F core outlet, 5 ppm oxygen level, 26 ft/sec sodium velocity, 316 stainless steel.

** Same basis as case 1 except for a < 2 ppm oxygen level.

*** Based on 800-1100°F average temperature, 1 ft/sec sodium velocity, Inconel-600.

Although stainless steel corrosion rates generally increase with increasing temperature, the average corrosion rate for the gas plenum region was reduced to take into account the "downstream effect." (This effect is the reduction in corrosion rate with increasing downstream position in isothermal, constant velocity flow channels.)

The release rate of each corrosion product nuclide is presented in Table 3 as a percentage of the stoichiometric corrosion rate for each nuclide. Deviations from 100% are adjustments made for reasons previously discussed.

Table 3
Isotopic Release Rate

Isotope →	<u>⁶⁰Co</u>	<u>⁵⁸Co</u>	<u>⁵⁴Mn</u>	<u>⁵¹Cr</u>	<u>¹⁸²Ta</u>	<u>⁵⁹Fe</u>
<u>Case 1:</u>						
Release Rate (% of stoich. rate)	20	10	200	100	1	100
<u>Case 2:</u>						
Release Rate (% of stoich rate)	80	40	200	100	1	100

Note that the absolute release rates of ⁶⁰Co and ⁵⁸Co are assumed to be independent of the oxygen level. (A reduction in the average corrosion rate is matched by an increase in the percentage of cobalt released.)

After these corrosion products are released to the coolant they are removed from the coolant by deposition on heat transport system surfaces and by radioactive decay. The total amount (or "inventory") of each of the corrosion product radionuclides in the coolant and deposits (corrected for decay) was calculated for both cases using the model described previously and in Appendix A. These inventories are presented in Table 4 for one, five, and ten years of reactor operation.

Table 4
HTS Corrosion Product Inventories*

		<u>Total HTS Inventory (Curies)</u>					
<u>Time</u>		<u>⁶⁰Co</u>	<u>⁵⁸Co</u>	<u>⁵⁴Mn</u>	<u>⁵¹Cr</u>	<u>¹⁸²Ta</u>	<u>⁵⁹Fe</u>
Case 1:	1 year	12.4	134	751	286	6.56	47.6
	5 years	48.3	138	1085	286	7.28	47.8
	10 years	73.3	138	1353	286	7.38	47.8

* Assumes 306 day fuel cycles.

Table 4 (Continued)

<u>Time</u>	<u>Total HTS Inventory (Curies)</u>					
	<u>⁶⁰Co</u>	<u>⁵⁸Co</u>	<u>⁵⁴Mn</u>	<u>⁵¹Cr</u>	<u>¹⁸²Ta</u>	<u>⁵⁹Fe</u>
Case 2: 1 year	12.4	134	188	71.4	1.64	11.91
5 years	48.3	138	271	71.4	1.82	11.95
10 years	73.3	138	338	71.4	1.84	11.95

After only one year of operation, all of the nuclides except ⁶⁰Co and ⁵⁴Mn have reached or almost reached their equilibrium inventory values. Longer lived ⁶⁰Co and ⁵⁴Mn continue to accumulate in the HTS cell throughout a 10 year operating period. In fact, the ⁶⁰Co inventory will not reach equilibrium for over 20 years.

For operation at 900°F core outlet, the corrosion rates in Table 2 (and hence the inventories in Table 4) would be reduced by a factor of seven. However, the above comments on the length of time necessary for each corrosion product nuclide to reach its equilibrium inventory are still valid.

4.1.2 Closed Loop Heat Transport System

In the closed loops, activated corrosion products are released from stainless steel surfaces inside the Closed Loop In-Reactor Assembly (CLIRA). These surfaces include the interior of the CLIRA itself and the fuel cladding, wire wraps and ducts of test subassemblies in the CLIRA. Since the CLIRA is a part of the FFTF core, it and its contents were divided into three zones, comparable to those of the core region, for

purposes of averaging corrosion rates and nuclear parameters (flux, cross sections, etc). For the closed loops Cases 1 and 2 were further expanded to include estimates for corrosion product sources from both driver fuel test sub-assemblies (lower temperature) and advanced fuel test assemblies (higher temperatures). CLS source information is summarized in Table 5.

Table 5
CLS Corrosion Product Source Parameters

<u>Source Region</u>	<u>Area (ft²)</u>	<u>Average Outlet Cladding Temp. (°F)</u>	<u>Avg. Corrosion Rate (mils/yr)</u>	
			<u>Case 1</u>	<u>Case 2</u>
Core (Fueled Zone)	12.4	*		
		1280	.14	.035
		1480	.70	.175
Axial Reflector	2.1			
		1280	.57	.14
		1480	2.8	.70
Gas Plenum	10.3			
		1280	.39	.10
		1480	2.0	.5

* The average temperature of fuel cladding at the outlet of a test assembly in a CLIRA.

The release rates of corrosion products to CLS primary sodium as percentages of stoichiometric corrosion rates for cases 1 and 2 are the same as those for the reactor primary system (see Table 3).

The total inventory of each of the corrosion product radionuclides in the CLS primary system after one and five years of operation is presented in Table 6.

Table 6

CLS Corrosion Product Inventories*

		<u>^{60}Co</u>	<u>^{58}Co</u>	<u>^{54}Mn</u>	<u>^{51}Cr</u>	<u>^{182}Ta</u>	<u>^{59}Fe</u>
Case 1:	1 year						
	1280°F	0.10	0.90	5.5	2.4	0.06	0.41
	1480°F	0.51	4.5	27.3	12.1	0.30	2.1
	5 years						
	1280°F	0.39	0.92	9.7	2.4	0.07	0.42
	1480°F	1.97	4.6	48.3	12.1	0.34	2.1
	1 year						
	1280°F	0.10	0.90	1.37	0.60	0.015	0.10
	1480°F	0.50	4.5	6.8	3.0	0.075	0.52
	5 years						
Case 2:	1280°F	0.39	0.92	2.4	0.60	0.017	0.10
	1480°F	1.96	4.6	12.1	3.0	0.085	0.52
	5 years						
	1280°F	0.39	0.92	2.4	0.60	0.017	0.10
	1480°F	1.96	4.6	12.1	3.0	0.085	0.52

* Assumes 306 day fuel cycles

Note that the cobalt inventories are assumed to be independent of oxygen level in the range 1-5 ppm. Again, with the exception of ^{60}Co and ^{54}Mn , all the nuclides reach their equilibrium inventories in about a year.

4.2 Distribution and Deposition of Corrosion Products

The corrosion products released to the reactor and closed loop primary coolants are transported by those coolants to all parts of the primary systems, where they deposit on pipe walls and component surfaces. Once deposited, the corrosion products become bonded to the steel surface and begin to diffuse into

the steel at rates approximating those predicted by metallic diffusion theory⁽¹⁾. While on the steel surface but before diffusion takes place the corrosion products can also exchange with both radioactive and non-radioactive corrosion products in the coolant and be transported further.

Obviously, the ultimate distribution of activated corrosion products in a primary sodium system is the result of a number of complex, and sometimes ill-understood phenomena.

However, experience gained from test loops (1) and operating reactors has provided a basis for predicting corrosion product distributions in sodium systems. The following general observations are applicable:

- Manganese preferentially migrates to and deposits in the cold legs and cold traps of sodium systems. Some manganese however, is deposited in the hot legs.
- Cobalt is not rapidly transported and therefore accumulates mostly in the hot legs of sodium systems.
- Deposition of all corrosion product species is increased in areas of turbulence (e.g. in pumps).

Little is known about the deposition of iron, chromium, or tantalum and so it was assumed that they deposit uniformly throughout a sodium system.

With these observations in mind, distribution patterns for each corrosion product isotope can be constructed for the

reactor and closed loop primary systems. The distribution of corrosion product inventories over the major components of those primary systems is presented in Tables 7 and 8. The distribution of each isotope is presented as percentages of the total inventory of each isotope released to the coolant (see Tables 4 and 6).

Table 7
HTS Corrosion Product Deposition Pattern

Major Component or Location	Percentage Distribution				
	⁶⁰ Co	⁵⁸ Co	⁵⁴ Mn	⁵¹ Cr	¹⁸² Ta ⁵⁹ Fe
Reactor Vessel	31	51	6.0	10.8	

(The following are for one HTS loop; multiply by 3 and add to the above figures to account for 100%)

Hot Legs:

Vessel-Pump	8.3	6.67	4.0	3.63
Pump-IHX	1.33	1.0	0.33	0.57
IHX	8.6	5.6	14.0	22.8
Cold Leg	0.67	0.33	8.33	1.53
Cold Traps	0.1	0.1	1.0	0.15
Pump	4.0	2.67	2.67	1.0

Table 8

CLS Corrosion Product Deposition Pattern

Location	Distribution (% of total inventory)				
	⁶⁰ Co	⁵⁸ Co	⁵⁴ Mn	⁵¹ Cr	¹⁸² Ta ⁵⁹ Fe
Supply Tank	3	2.5	8	6	
Drain Tank	3	2.5	8	6	
Surge Tank	3	2.5	8	6	
Cold Trap	5	5	10	6	
IHX	15	15	13	20	
Smear	3	2.5	8	6	
Piping	25	20	25	30	
CLIRA	43	50	20	20	

The "smear" refers to the minor components and piping inside a CLS module which were homogenized into a rectangular parallel piped for radiation dose calculations.

These distributions and the corrosion product inventories developed in the previous section were used with geometric models of the reactor and closed loop primary systems to calculate radiation doses at points in and near an HTS cell and a CLS module. The results of these calculations are presented below.

4.3 Radiation Levels

Geometric models of an HTS cell and a CLS module were used in conjunction with the above inventory and distribution information to calculate the radiation doses from corrosion products at nine selected points in an HTS cell and six points near a CLS module (see Appendix B for details). These locations, or dose points, were chosen for their importance in maintenance operations and/or technical interest.

4.3.1 HTS Radiation Levels

The nine dose points selected for the HTS cell radiation level calculations are shown in Figure 1. All points are in the plane of the piping except points 8 and 9 which are respectively 2 and 10 feet above the check valve. (Check valve maintenance will be conducted from positions above the valve.) Points 1-7 are located two

feet from the component except for point 4 which is midway between the cold leg and hot leg piping, or about 21 inches from either pipe.

The system was assumed to be drained of sodium. The activity of any residual film of sodium was assumed to have decayed away.

No shadowing effects (i.e. one component shielding another) were included in the calculation. Although this would make some difference in the dose at a point, it would not be expected to appreciably affect the whole body dose upon which access is based, as complete shadowing of the whole body is unlikely.

The primary contributing activated corrosion products are ^{60}Co , ^{58}Co , and ^{54}Mn . Total dose rates due to the isotopes are presented for Cases 1 and 2 in Table 9. Contributions from ^{51}Cr , ^{59}Fe , and ^{182}Ta turn out to be minor except for very short operating times (a few months). Both ^{51}Cr and ^{59}Fe reach equilibrium quickly due to short half lives. The calculated ^{51}Cr dose rate at point 7 near the check valve for Case 1 was 0.03 R/hr, truly negligible. The ^{59}Fe dose rate was always less than 5% of the total, thus it could also be considered unimportant.

Figure 1. HTS Cell Layout and Dose Points

Table 9
Corrosion Product Radiation Levels in
an HTS Cell

(R/hr)

<u>Dose Point</u>	<u>Case 1</u>			<u>Case 2</u>		
	<u>1 yr</u>	<u>5 yr</u>	<u>10 yr</u>	<u>1 yr</u>	<u>5 yr</u>	<u>10 yr</u>
1	2.45	3.67	4.61	1.04	1.64	2.07
2	2.60	3.88	4.87	1.10	1.72	2.17
3	3.32	4.94	6.19	1.20	1.88	2.38
4	8.18	12.01	15.02	2.65	4.03	5.07
5	5.61	8.19	10.24	1.59	2.39	3.01
6	3.41	5.12	6.42	1.51	2.38	3.00
7	9.33	13.64	17.06	2.68	4.03	5.07
8	6.27	9.16	11.44	1.77	2.69	3.38
9	1.49	2.19	2.74	0.47	0.72	0.91

Dose Rates From Minor Contributors

7 (^{182}Ta)	0.05	0.05	0.05	0.01	0.01	0.01
7 (^{51}Cr)	0.03	0.03	0.03	0.01	0.01	0.01
7 (^{59}Fe)	0.38	0.38	0.38	0.09	0.09	0.09

^{182}Ta produces gamma radiation as penetrating as ^{60}Co but at lower dose rates for the same inventory. Since the calculated ^{182}Ta inventories are well below those for ^{60}Co (Table 4) for all but the shortest operating times, the associated ^{182}Ta dose rate can be ignored. However, should stoichiometric corrosion of ^{182}Ta occur, appreciable inventories of ^{182}Ta would be present. During the first few months of operation, and assuming stoichiometric

corrosion, the ^{182}Ta contribution might increase the total dose rates by factors of 2 or 3 over those presented here.

The most important thing to note about the predicted dose rates presented in Table 9 is that, for either Case, they rapidly exceed 1 R/hr for all dose points. This level of radiation can cause access problems for maintenance work in the HTS cells. As a recent HTS Engineering Maintenance Study (2) shows, 200 mR/hr must be regarded as an upper limit for radiation exposure because "it is administratively impractical to perform maintenance work in radiation fields of 200 mR/hr or greater. The basic reason being the inefficient use of manpower due to the possible distance in a radiation zone which must be traversed to get to the work location." Radiation levels predicted by Case 1 exceed 200 mR/hr by at least a factor of 50; Case 2 levels generally exceed 200 mR/hr by a factor of ten.

Some caution should be exercised when comparing the results of Cases 1 and 2. The reduction in radiation levels from Case 1 to Case 2 is based upon meager evidence. It has been observed that corrosion of cobalt base alloys in sodium is less than that of stainless steel and not dependent upon the oxygen content of sodium⁽⁴⁾. Manganese reacts strongly

with oxygen (the free energy of formation of MnO is more negative than that of Na₂O or CoO) so that the release rate of manganese is probably dependent on the oxygen level in the sodium. Direct evidence to support those conclusions is not now in existence; experiments are being run at HEDL.

In addition, it has recently been found that oxygen levels in sodium systems are much lower than previously thought, and that oxygen levels previously reported at 5 ppm were really about 1 or 2 ppm. Thus, the radiation levels calculated for Case 1 (based upon data taken at what was thought to be 5 ppm) may really be applicable for Case 2 (1 or 2 ppm). This, of course, is a disappointment. However, reducing the oxygen level from a true 1 or 2 ppm to less than 1 ppm may effect the same type of reduction in radiation levels. Experimental investigation of this effect is currently being conducted.

Selective shielding of components in an HTS cell was also examined. The selective shielding of one or a few components can have only a limited effect because of the wide distribution of pertinent dose points and because many components contribute to the total dose rate at any one dose point. As an example of the limits involved, selectively shielding the IHX to eliminate its contribution to the dose rate was considered. Dose rates near the IHX are among the

highest in the HTS cell and the IHX contains about 70% of the deposition area in the HTS cell making it a prime candidate for selective shielding. The percentage contribution to each of the isotopic dose rates from the IHX is shown in Table 10.

Table 10
IHX Dose Rate Contributions

<u>Point</u>	<u>^{60}Co</u>	<u>^{58}Co</u>	<u>^{54}Mn</u>
1	4%	3%	8%
2	4	2	7
3	25	16	28
4	9	6	7
5	38	30	13
6	3	2	5
7	37	28	14
8	26	18	9
9	9	3	4

Only for points near the IHX is the contribution significant, and even there it is limited to about one third of the total. Using these percentages, the actual reduction in the total dose rate due to a 100% shielded IHX after ten years of operation was calculated. Results are shown in Table 11. Isotopic contributions were taken from Table 9 (Case 1). No major reductions in any of the dose rates occurred.

Table 11
Dose Rates For Shielded and Unshielded IHX

<u>Point</u>	<u>Unshielded Dose Rate</u>	<u>Dose Rate Without IHX</u>
1	4.61 R/hr	4.30 R/hr
2	4.87	4.57
3	6.19	4.51
4	15.02	14.01
5	10.24	8.75
6	6.42	6.13
7	17.06	14.49
8	11.44	10.35
9	2.74	2.62

Only general shielding of all of the major primary system piping and components will significantly reduce the HTS cell radiation levels from the corrosion product inventories predicted here. However, a reduction of the inventories of ^{60}Co , ^{58}Co , and ^{54}Mn would cause a concomitant reduction in radiation levels. Efforts to evaluate several means of reducing the corrosion product inventories in the FFTF are currently being made as a part of HEDL's Radiation and Repair Engineering Program⁽³⁾.

4.3.2 CLS Radiation Levels

The magnitude of the radiation field due to activated corrosion products was calculated for five selected dose points in and around a closed loop module. These dose

points were at locations where maintenance activities are likely and are shown in Figure 2. In addition, the radiation field two feet from a bare 3" pipe was also calculated, and is assumed to be representative of the field around the interconnecting piping between a CLIRA and a closed loop module.

Cases 1 and 2 were evaluated for two sets of corrosion rates (Table 5), one typical of driver fuel and one typical of advanced fuel which might be tested in the closed loops. Radiation levels for the six CLS dose points are presented in Table 12.

Table 12
CLS Corrosion Product Radiation Levels

<u>Dose Point</u>	<u>R/hr</u>		<u>R/hr</u>	
	<u>1280°F Outlet</u>		<u>1480°F Outlet</u>	
	<u>1 yr</u>	<u>5 yrs</u>	<u>1 yr</u>	<u>5 yrs</u>
<u>Case 1:</u>				
1	.15	.25	.73	1.24
2	.11	.18	.53	.91
3	.12	.20	.58	.98
4	.10	.17	.49	.83
5	.12	.20	.59	1.00
6 (3" pipe)	.037	.045	.13	.23

Figure 2. Closed Loop Module Dose Points and Component ARRGT

<u>Dose Point</u>	R/hr		R/hr	
	1280°F Outlet		1480°F Outlet	
	<u>1 yr</u>	<u>5 yrs</u>	<u>1 yr</u>	<u>5 yrs</u>
Case 2:				
1	.044	.076	.22	.38
2	.033	.056	.16	.28
3	.038	.065	.19	.33
4	.031	.053	.15	.26
5	.039	.066	.19	.33
6	.010	.016	.048	.082

Operation of the closed loops with FTR driver fuel at a 1280°F outlet sodium temperature does not appear to produce corrosion product radiation levels which would seriously interfere with contact maintenance. Only for higher temperature fuel tests could radiation from corrosion product deposits interfere with contact maintenance of closed loop primary components. Even at the elevated temperatures, the predicted CLS radiation levels are considerably below those predicted for an HTS cell. Again, caution should be exercised in comparing Case 1 and Case 2 oxygen level assumptions to those predicted for FFTF (See previous section).

To evaluate the sensitivity of the dose point radiation levels to the distribution of the corrosion products around a CLS module, a parametric study was made. An extra quantity of corrosion products was assigned, in turn, to each of the major closed loop components. This "extra" quantity of corrosion products was "taken" from the inventory assigned to the CLIRA (see Table 8) and equaled 20% of the total inventory of each corrosion product isotope. Except for

the component with the extra deposits, all of the other components had the same corrosion product inventories that were used to calculate the radiation levels in Table 12. The Case 2 high temperature inventories at 5 years were used for this parametric study and the results are presented in Table 13.

Table 13
Effect of Distribution on Dose Point
Radiation Levels

<u>Location of</u> <u>"Extra" Deposit</u>	<u>Dose Point Radiation Level (R/hr)</u>					
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>
Tanks	.73	.49	.43	.39	.46	.082
Cold Trap	.41	.34	.47	.42	.49	.082
IHX	.39	.30	.48	.32	.48	.082
"Smear"	.66	.57	.61	.49	.56	.082
- - - - -						
Levels from Table 12 (+ 20%)	.46	.34	.40	.31	.40	.082

For comparison, the radiation levels from Table 12 were increased by 20%, the amount of the extra inventory. With the exception of the case in which most of the corrosion products are assigned to the "smear," the dose point radiation levels appear to be relatively insensitive to the distribution pattern of the corrosion products. The explanation for this is contained in the geometry of the situation. The assumed dose points are located on an external face and the top of the module. Radiation from separate

sources in the interior of the module tends to "blend" and lose its source identity by the time it reaches the exterior dose points. Therefore, the distribution of corrosion products over the components within the module (i.e., IHX, tanks, cold trap) does not greatly affect the dose point radiation levels. The exception to this occurs when a dose point is adjacent to a component, such as point #1 at the bottom of the module adjacent to Tank T-1. Assigning most of the corrosion products to the tanks significantly increases the radiation levels at point #1. It does not materially affect the radiation levels at the other dose points.

The sensitivity of the dose point radiation levels to the amount of corrosion products assigned to the smear is a result of the proximity of the dose points to the "source." For this model, the smear is a rectangular parallelepiped of the same dimensions as the closed loop module. The dose points are one foot from the surface of this "component" and thus are quite sensitive to the corrosion product inventory assigned to the smear.

The conclusion one can draw from the analysis of the effects of the corrosion product distribution pattern on the dose point radiation levels is that the radiation levels external to the module are relatively insensitive to the distribution pattern of corrosion products within the module. Thus, the only critical distribution parameter is the total

corrosion product inventory in a closed loop module, and not its distribution within the module. This is important when considering the merits of shielding specific components within the module. Unless one component contains most of the corrosion products, shielding it is not likely to be very effective.

This is demonstrated in Table 14, which presents dose point radiation levels for the same operating conditions, inventories, and distribution patterns used in Table 13, but with a 100% effectively shielded cold trap.

Table 14
Effect of Cold Trap Shielding

<u>Location of "Extra"Deposit</u>	<u>Dose Point Radiation Levels (R/hr)</u>					
	<u>1</u>	<u>2</u>	<u>3</u>	<u>4</u>	<u>5</u>	<u>6</u>
Tanks	.72	.47	.38	.34	.40	.082
Cold Trap	.37	.26	.27	.21	.27	.082
IHX	.37	.27	.43	.26	.42	.082
"Smear"	.65	.54	.56	.43	.50	.082
- - - - -						
Levels from Table 12 (+20%)	.46	.34	.40	.31	.40	.082

Compare Tables 13 and 14. As expected, shielding the cold trap significantly reduces the dose point radiation levels only when a large fraction of the corrosion products are assumed to deposit in the cold trap. Even then, the dose point radiation levels are not greatly reduced.

It should be remembered that quantitative information on the distribution of corrosion products in sodium heat transport systems is nearly non-existent. Thus, selectively shielding one or more individual components within a closed loop module may or may not reduce the general corrosion product radiation levels.

5.0 CONCLUSIONS

The first conclusions that can be drawn from this work are that the radiation levels due to activated corrosion products will probably be in the range of 1-15 R/hr for points within an HTS cell and within a range of 30-1000 mR/hr near a CLS module after five years of reactor operation at ultimate conditions. The corrosion product radiation levels in an HTS cell are predicted to be high enough to interfere with contact maintenance. The radiation levels near a CLS module may or may not affect a contact maintenance philosophy.

Another important result of this work concerns the indication that reducing the oxygen levels in the primary sodium of the HTS and CLS can have a significant effect on corrosion product radiation levels. The assumption has been made in this report that a reduction in oxygen level will effect a reduction in radiation levels by decreasing the release of ^{54}Mn (and ^{51}Cr , ^{59}Fe , ^{182}Ta). Whether or not this is true, the range of oxygen levels over which it would be true, and which radionuclide release rates are dependent upon oxygen level still remain to be determined.

The test systems being built at HEDL will investigate the effect of oxygen level in the 0.1 to 5 ppm range. They will be equipped with sensitive devices for measuring oxygen level (electrochemical oxygen meter, vanadium wire device) so that the question as to true oxygen levels mentioned in section 4.3.1 can be resolved.

Selective shielding of one major component (IHx) in an HTS cell was shown to be an ineffective means of reducing the general radiation level in the cell. This is because of the wide distribution of dose points and because many components contribute to the total dose rate at any one dose point. Only general shielding of all of the major primary system piping and components will significantly reduce the corrosion product radiation levels in an HTS cell.

For the CLS module, it can be concluded that radiation levels external to the module are relatively insensitive to the distribution pattern of corrosion products within the module. This is an important consideration when attempting to decide the merits of shielding specific components within a module. Unless one component contains most of the corrosion products (an unlikely event), shielding that component is not likely to be effective.

Finally, tests of FTR driver fuel in the closed loops with outlet sodium temperatures below 1280°F are predicted to produce corrosion product radiation levels no higher than 250 mR/hr after

5 years operation for either Case 1 or 2. This would probably not interfere with contact maintenance on the module; only operation with high temperature fuel will produce corrosion product radiation levels which could interfere with contact maintenance.

6.0 REFERENCES

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3. "Radiation and Repair Engineering Program Plan," July 1971, prepared by HEDL, W/FFTF 713040.
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APPENDIX A
CALCULATIONAL MODEL FOR CORROSION PRODUCT
INVENTORIES IN AN HTS CELL AND A CLS MODULE

The inventories of activated corrosion products in the reactor and closed loop primary systems were calculated by using a mass balance technique.

The change in the inventory of any radionuclide in the primary system is given by

$$\frac{dQ}{dt} = \left(\begin{array}{l} \text{release rate} \\ \text{from core} \end{array} - \text{radioactive decay} \right) = RA - \lambda Q \quad (A-1)$$

where Q = primary system radionuclide inventory (curies)

λ = decay constant = $\ln 2 / \text{half-life}$ (time⁻¹)

A = area of corrosion product nuclide source (area)

R = average release rate of nuclide (curies/area-time)

The release rate of each radionuclide is a function of its concentration in the material being released and the rate of release (corrosion rate) of that material.

The buildup of radioactive atoms produced by irradiation of a target material is given by

$$\frac{dN^*}{dt} = \phi \sigma N - \lambda N^* \quad (A-2)$$

where N^* = number of radioactive atoms

N = number of target atoms

ϕ = average neutron flux (n/cm²-sec)

σ = flux averaged activation cross section (barns)

λ = decay constant (sec⁻¹)

The solution to Equation(A-2) is

$$N^* = \frac{\phi \sigma N}{\lambda} (1 - e^{-\lambda t}) , \quad (A-3)$$

if $N^* = 0$ at $t = 0$. This is certainly true at reactor startup and is nearly true at the beginning of each fuel cycle since most of the corrosion products originate in the fuel subassembly surfaces, which are unirradiated at the start of a fuel cycle. If the stainless steel alloy under consideration has a fraction f_c of target element and f_i of target isotope in the target element, then the isotopic concentration of the radionuclide in the steel (and hence the corroding material) is given by

$$X = \frac{N^*}{N} f_i f_c \frac{\rho}{M} \frac{N_0}{D} , \text{ (curies/cm}^3\text{)} , \quad (A-4)$$

where N_0 = Avogadro's Number

ρ = alloy density, gms/cm³

M = alloy molecular weight

D = conversion factor, 3.7×10^{10} disintegrations/sec-curie.

Combining Equations (A-3) and A-4) yields

$$X = \frac{\sigma \phi f_i f_c \rho N_0}{M D} (1 - e^{-\lambda t}) . \quad (A-5)$$

The release rate (R) at any given time is equal to the product of the bulk corrosion rate of the stainless steel (C, in cm³/area-time) and the radionuclide concentration of the steel (X). Over the lifetime of a reactor system, many fuel

cycles are completed. During each cycle, the amount of radioactive corrosion products released from the irradiated steel (which is proportional to X) increases with time [Equation (A-5)]. After each cycle, the reactor is refueled and the process is repeated. If one assumes $X = 0$ at the beginning of each fuel cycle, then the average value of X over one fuel cycle is

$$\begin{aligned}\bar{X} &= \int_0^{T_c} K (1 - e^{-\lambda t}) dt / \int_0^{T_c} dt \\ &= K \left[1 + \frac{e^{-\lambda T_c}}{\lambda T_c} - \frac{1}{\lambda T_c} \right]\end{aligned}\quad (A-6)$$

where T_c = length of one fuel cycle, (time)

K = collection of constants from Eq. (A-5), (curies/cm³).

If one assumes a constant corrosion rate (C) and uses the average value for the concentration of a radionuclide of interest ($\bar{X}$), then Equation (A-1) becomes

$$\frac{dQ}{dt} = \bar{X} C A - \lambda Q, \quad (A-7)$$

and, for $Q = 0$ when $t = 0$,

$$Q = \frac{\bar{X} C A}{\lambda} (1 - e^{-\lambda t})$$

$$\text{or } Q = \frac{\sigma \phi f_i f_c \rho N_o C A}{M D \lambda} \left[1 + \frac{e^{-\lambda T_c}}{\lambda T_c} - \frac{1}{\lambda T_c} \right] [1 - e^{-\lambda t}]. \quad (A-8)$$

This expression was used to calculate the inventories of activated corrosion products as functions of time in the reactor and closed loop primary heat transport systems.

Since the temperature, sodium velocity, and nuclear parameters vary throughout the core region, this source region was divided into four zones, and average corrosion rates, fluxes, and activation cross-sections were assigned to each zone. The core zone consisted of the fueled length of the subassemblies; the axial reflector zone was six inches high and situated directly above the core zone; the gas plenum zone extended 42 inches above the axial reflector zone; and the radial reflector zone consisted of four-foot lengths of the reflector subassemblies which were located adjacent to the core and axial reflector zones. The corrosion rates and stainless steel surface areas for each zone were presented in Table 2; the nuclear parameters for each zone are presented in Table A-1.

Table A-1
Nuclear Parameters*

<u>Parameter</u>	<u>Zone</u>			
	<u>Core</u>	<u>Axial Reflector</u>	<u>Gas Plenum</u>	<u>Radial Reflector</u>
Average Neutron Flux (n/cm ² -sec)	4.2x10 ⁺¹⁵	1.87x10 ⁺¹⁵	2.92x10 ⁺¹⁴	8.89x10 ⁺¹⁴
Flux Averaged Activation Cross-Sections (Barns)				
⁵⁹ Co (n, γ) ⁶⁰ Co	.122	1.730	3.470	1.949

* Supplied by Radiation and Shielding Analysis, HEDL, 4/29/71.

Table A-1 (Continued)

Parameter	Zone			
	Core	Axial Reflector	Gas Plenum	Radial Reflector
Flux Averaged Activation Cross-Sections (Barns) (Continued)				
$^{58}\text{Ni}(n,p) \ ^{58}\text{Co}$	.014	.004	.001	.003
$^{54}\text{Fe}(n,p) \ ^{54}\text{Mn}$	.009	.003	--	.002
$^{60}\text{Ni}(n,p) \ ^{60}\text{Co}$	.001	--	--	Not determined
$^{58}\text{Fe}(n,\gamma) \ ^{59}\text{Fe}$	.012	.042	.070	"
$^{50}\text{Cr}(n,\gamma) \ ^{51}\text{Cr}$	.035	.086	.150	"
$^{181}\text{Ta}(n,\gamma) \ ^{182}\text{Ta}$	1.120	9.180	19.700	"

Note the large cross sections for the $^{59}\text{Co}(n,\gamma) \ ^{60}\text{Co}$ and $^{181}\text{Ta}(n,\gamma) \ ^{182}\text{Ta}$ in the axial reflector and gas plenum zones. The flux softens markedly above the fueled region of the core and it is in this area where most of the ^{60}Co and ^{182}Ta are produced. The other activated corrosion products are produced primarily in the core zone.

For the closed loop system analysis, the CLIRA and its contents were divided into zones corresponding to those of the driver core. It was assumed that the nuclear parameters for the CLIRA and contents were the same as those of the driver core.

To calculate the inventory of any corrosion product nuclide, Equation (A-8) was applied to each zone and the resulting inventories summed to yield a total. This total was then used with an assumed deposition pattern to calculate dose rates near primary system components. The dose rate calculations are described in Appendix B.

APPENDIX B
CALCULATION OF SPECIFIC DOSE RATES

Sources in both the HTS cell and closed loop module are contained in a geometrically complex maze of piping and components. The ISOSHLD⁽¹⁾ point kernel code, used to calculate all dose rates, allows only a few simple source geometries. It was thus necessary to break each of these systems down to a number of components each one of which could be handled by a particular ISOSHLD geometry.

HTS Cell Model

The HTS cell source was modeled by a set of nineteen right circular cylinders representing individual pipe runs, the primary pump, the intermediate heat exchanger, and the check valve. These components are shown and identified by number in Figure B-1. The coordinates of the end points of each cylinder axis are shown in Table B-1. The configuration and internal composition of each component are shown in Table B-2.

The locations of the nine selected dose points have been previously described (Figure 1). The coordinates of each are given in Table B-3. For each of these dose points, the dose rate due to one curie of the isotope of interest was calculated. Folding these specific dose rates with the individual component nuclide inventories then provides the total dose at that dose point. No shadowing effects (one component shielding another)

Figure B-1. HTS Cell Geometrical Model

Table B-1
HTS Component Locations*

<u>Component</u>	<u>Identification</u>	<u>Coordinates, feet</u>					
		<u>X₁</u>	<u>Y₁</u>	<u>Z₁</u>	<u>X₂</u>	<u>Y₂</u>	<u>Z₂</u>
1	Hot Leg Piping	11.5	5.0	530.8	39.5	5.0	530.7
2	Hot Leg Piping	39.5	5.0	530.7	39.5	34.3	530.6
3	Hot Leg Piping	39.5	34.3	530.6	32.4	34.3	530.6
4	Hot Leg Piping	32.4	34.3	530.6	32.4	12.5	530.5
5	Hot Leg Piping	32.4	12.5	530.5	24.5	12.5	530.5
6	Hot Leg Piping	24.5	12.5	530.5	24.5	12.5	512.5
7	Hot Leg Piping	21.2	16.3	506.0	21.2	16.3	530.2
8	Hot Leg Piping	21.2	16.3	530.2	13.5	23.3	530.2
9	Hot Leg Piping	13.5	23.3	530.2	15.0	25.3	530.2
10	Cold Leg Piping	21.3	27.6	506.0	21.3	27.6	529.6
11	Cold Leg Piping	21.3	27.6	529.6	23.4	25.4	529.6
12	Cold Leg Piping	23.4	25.4	529.6	23.4	18.0	529.9
13	Cold Leg Piping	23.4	18.0	529.9	27.1	18.0	529.9
14	Cold Leg Piping	27.1	18.0	529.9	27.1	34.3	530.3
15	Cold Leg Piping	27.1	34.3	530.3	25.4	34.3	530.3
16	Cold Leg Piping	19.7	34.3	530.5	10.0	34.3	530.8
17	Pump	17.0	12.7	507.5	17.0	12.7	527.0
18	IHX	17.0	27.6	513.6	17.0	27.6	533.5
19	Check Value	25.4	34.3	530.3	19.7	34.3	530.5

* - Coordinates of cylinder axis endpoints.

Table B-2

HTS Cell Component Descriptions

<u>Component (s)</u>	<u>Configuration and Composition</u>
1 thru 6	28" O.D. pipe, 3/8" wall, source uniformly distributed in pipe interior, sodium drained, source region at air density.
7 thru 16	16" O.D. pipe, other specifications as above.
17	Primary pump, homogenized source region 6 1/2ft in diameter containing 5 volume percent steel, surrounded by 3 in. of steel and 7 in. of 11.0 lb/ft ³ insulation (assumed same percentage composition as ordinary concrete), sodium drained from source region.
18	Intermediate heat exchanger, homogenized source region 6 1/3 ft. in diameter containing 3 volume percent steel and no sodium, surrounded by 1.2 inches of steel.
19	Check valve, homogenized source region 30.5 inches in diameter containing air only, 3/4" wall.

Table B-3

Dose Point Locations

<u>Point</u>	<u>X,ft.</u>	<u>Y,ft.</u>	<u>Z,ft.</u>
1	22.0	1.8	530.7
2	42.7	20.0	530.6
3	13.9	19.3	530.2
4	29.5	25.0	530.1
5	18.0	37.0	530.7
6	36.3	8.2	530.7
7	22.6	31.0	530.4
8	22.6	34.3	533.7
9	22.6	34.3	541.7

were included in the calculation. Gamma source strengths and energies were taken from Reference 2.

For the cylindrical source, ISOSHL D has two geometry options, a dose point off the end but on the centerline and a dose point off the side but between the endpoints. For those dose points far off the centerline and also beyond an endpoint, as shown by points P_1 , and P_2 in Figure 2a, the following methods are used. For point P_1 , the dose rate contribution from a source cylinder of length, l , is subtracted from that due to one of length, h (see Figure B-2b). Point P_2 represents points for which the above method can suffer from loss of accuracy caused by subtracting two numbers of about the same magnitude. On the basis of trial runs for configurations typical of this study, it was determined that the above method was adequate for points for which $\alpha < 45^\circ$ (see Figure B-2a). For those points, P_2 , for which $\alpha > 45^\circ$, the alternate method shown in Figure B-2c was used. The source cylinder was rotated about the near end to position A and the dose rate calculated for this configuration. This method over-estimates the dose rate from that component, but in every case in which this method was necessary, it was determined that there was no appreciable effect on the total dose, which includes contributions from all other components.

Figure B-2. Dose Points Beyond a Cylinder End

Closed Loop Module Model

The closed loop module is an even more complex maze of piping and components. In the simplified model used in the calculations, the closed loop module was assumed to consist of five major components with the rest of the components and the piping homogenized into a rectangular parallelepiped. Positions and identification of the components are shown in Table B-4. Coordinates given are those for the axis ends of components modeled as cylinders and for the diagonal ends of the parallelepiped "smear".

Table B-4

Closed Loop Component Identification and Location

<u>Component</u>	<u>Identification</u>	<u>Model</u>	<u>Coordinates, feet</u>					
			<u>X₁</u>	<u>Y₁</u>	<u>Z₁</u>	<u>X₂</u>	<u>Y₂</u>	<u>Z₂</u>
T-1	Supply Tank	Cylinder	4.2	0.7	521.	4.2	9.0	521.
T-2	Drain Tank	Cylinder	9.0	0.7	521.	9.0	9.0	521.
T-4	Pri. Surge Tank	Cylinder	9.8	4.8	534.	9.8	4.8	541.
CT-1	Cold Trap	Cylinder	10.0	8.0	535.	10.0	8.0	539.
E-1	Heat Exch.	Cylinder	7.7	2.5	534.	7.7	2.5	543.
SMEAR	See Text	Rect. Parallelopiped	2.0	0.5	518.5	12.0	9.5	545.5

The five major components were modeled as right circular cylindrical tanks with the characteristics given in Table B-5. Each was surrounded with four inches of 20 lb/ft³ insulation. Internal composition of each component is also shown.

Table B-5

Closed Loop Component Description

<u>Component</u>	<u>Internal Radius, inches</u>	<u>Wall Thickness, inches</u>	<u>Internal Composition</u>
T-1	17.5	0.5	Air
T-2	23.5	0.5	Air
T-4	12.5	0.5	Air
CT-1	11.5	0.5	15 lb/ft ³ steel
E-1	13.88	0.625	34.4 lb/ft ³ steel
SMEAR	-----	-----	5.66lb/ft ³ steel, 3.33lb/ft ³ insulation

Approximate locations of the five selected dose points were previously shown in Figure 2. Coordinates are given in Table B-6. Points 4 and 5 are above, respectively, the cold trap and the heat exchanger. Points 1, 2, and 3 are on the module vertical mid plane.

Table B-6

Closed Loop Dose Point Locations

<u>Point</u>	<u>Coordinates, feet</u>		
	<u>X</u>	<u>Y</u>	<u>Z</u>
1	1.0	4.8	524.
2	1.0	4.8	527.
3	1.0	4.8	540.
4	10.0	8.0	546.5
5	7.7	2.5	546.5

The geometrical treatment of the major components, all cylinders, was the same as given above for the HTS Cell Components, with the following exception. Dose points 4 and 5 are on the centerlines above, respectively, the cold trap and intermediate heat exchanger. The ENDCYL option (dose point on centerline, off cylinder end) in ISOSHLD was used for these two cases. The ISOSHLD option for a rectangular parallelepiped allowed calculation of the dose on the centerline only. As the dose rate near the surface would not be expected to decrease appreciably except near an edge, this conservative value was used for all dose points on a particular side.

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