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RADIOACTIVITY
IN FFTF CLOSED LOOPS

W. F. Brehm, P. L. Koehmstedt,
E. A. Kovacevich, and D. W. Shannon

OCTOBER 1967

AEC RESEARCH &
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October 1967

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RADIOACTIVITY IN FFTF CLOSED LOOPS

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INTRODUCTION

The first phase of the development program for FFTF contamination control is a series of laboratory scale studies of sodium-fission product interactions at PNL (Tasks RS-8 and RS-9) and other related tasks at LASL, AI, BNL, and GE, (PNL is presently preparing further plans for DRDT consideration of redirection of basic fission product studies at LASL). The present FFTF Development Plan calls for pilot plant scale testing those radioisotope control concepts selected by these laboratory screening studies for probable use on FFTF.

The second phase for development and testing on an engineering scale requires construction of a scaled model of an FFTF closed loop in a shielded cell capable of handling fully irradiated fuel material. This Engineering Scale Loop and the associated program are described in detail in FFTF Task IT-9. The major goals of Task IT-9 are: (1) to develop and proof test FFTF closed loop fission product control systems (2) to determine system design needed to run fuel to failure and recover, and (3) to develop and test FFTF closed loop operating and maintenance procedures after fuel failure. During FY-68 the task is funded at \$40,000 to prepare the design of the Engineering Scale Loop for fully irradiated defect fuel tests and to initiate procurement of long lead items for this loop. The estimate for the total capital cost of the facility is \$400,000 to \$500,000.

Since the FY-68 189 submittals further discussion between PNL and DRDT of the role of Task IT-9 in the FFTF program has indicated the need for the following quantitative study of the expected levels of deposited radioisotopes in the FFTF system. This study (1) discusses relevant FFTF requirements, (2) describes in quantitative terms probable FFTF closed loop contamination levels from transported activated corrosion products, vented fuel, and failed fuel, (3) relates potential FFTF radiation levels to past sodium reactor experience, and to FFTF design problems, and (4) relates IT-9 development activities and expected accomplishments to FFTF needs.

SUMMARY AND CONCLUSIONS

Two major FFTF design requirements are the capability of (1) running closed loop fuel tests to failure and (2) at the same time maintaining a system of high reliability and availability.

Radioactivity is deposited in the piping from activated stainless steel corrosion products (Co^{58} , Mn^{54}) and fission products released from vented or failed fuel tests. Analysis of the contamination levels indicates that although radiation from activated corrosion products will be serious, major fission product release from fuel failure will produce radiation levels 100 to 1000 times higher than either corrosion product transfer or vented fuel tests. It is a corollary of the high neutron flux of FFTF that a high fission product inventory in the fuel results. It would appear that release of 1% of the fission products in one FFTF subassembly with only 11 days burnup will produce a fission product release 10 times higher than the fission product release in the 1959 accident at SRE. Loss of 100% of one subassembly releases 1000 times as much fission products (10^7 Ci) as occurred at SRE (10^4 Ci). The release of the fission products in one FFTF pin (4×10^4 Ci) produces a contamination level comparable to the Na^{24} inventory at shutdown. These contamination levels are sufficiently high to cause concern for the consequences of driver fuel failure in the main FTR core as well as the closed loops.

It is probable that within a few days or weeks after a major fission product release, most of the contamination will be "plated out" and possibly alloyed with the piping since "plate out" was observed in all sodium reactor fuel failures to date (SRE, LAMPRE, BR-5 and Fermi). Decontamination of a sodium system is inherently different from decontamination of an aqueous reactor. Decontamination of an aqueous reactor involves removing a surface oxide film on the stainless steel whereas decontamination of sodium systems involves removing the stainless steel surface. It would be best to prevent fission product deposition or achieve controlled deposition in a fission product trap and avoid the need for decontamination. However, the Russians were fairly successful in decontaminating the BR-5 and achieved decontamination factors of over 200.

of operating and fabrication variables with the ultimate goal of developing reliable and economic FBR fuel.

Studies of FFTF functional testing requirements will be completed to aid in determining the specific number, sizes, and operating characteristics of closed loops and specific tests.

- (3.3.5) Test fuels to maximum limits, including tests up to failure. Although meltdown testing is precluded from closed-loop requirements, recovery from accidental meltdown is to be included in the design.
- (3.3.6) Operate with fission products in coolant to allow testing of vented fuel types or to determine effects of operation after fuel failure.
- (3.3.15) Equipment cells should be arranged for maintenance considering alpha and gamma radiations.
- (3.3.16) Provisions are required for cleaning up both cover gas and sodium including fission product removal, in-line instrumentation, and sampling for determining purity.

Page 5-1 (5.0)

Maintenance Philosophy - The order of preference of performing maintenance work at FFTF is as follows:

- (A) Contact maintenance - repair of components in place to the extent permissible, consistent with radiation exposure and other personnel safety considerations.
- (B) Removal of components for decontamination and repair by contact maintenance.
- (C) Decontamination, either internal, external or both, of components and cells to permit either A or B above, or for remote repair in maintenance cell.
- (D) Remote maintenance - in those areas in which irradiation or excessive contamination

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Analysis of several alternatives for maintenance of closed loop equipment indicates that making provisions for remote replacement of a limited number of equipment modules is desirable with provisions for major module decontamination and maintenance outside the closed loop cell. However, by optimum placement of shielding inside the cell, and segregating hot components it is probable that personnel entry for contact maintenance can still be done in many parts of the cell except after worst case accidents. Significant improvements in availability will ensue if a module does not have to be replaced for every minor failure of a wire, thermocouple, or valve.

The present development plan to obtain information and test the design adequacy of contamination control methods incorporates tracer level experiments at AI, LASL, BNL and PNL (Tasks RS-8 and RS-9). We hope laboratory scale studies will provide sufficient data for the prediction of deposition patterns of activated corrosion products. These laboratory studies are also essential to understanding necessary to predict what will happen after a fuel failure. However, the laboratory studies can never really come to grips with the engineering scale problems of fuel failure because fully irradiated mixed oxide has different characteristics from tracer level irradiated fuel. Since many of the rupture debris deposition patterns will be flow or geometry sensitive, it is dangerous to extrapolate from bench scale tests to full reactor scale. The IT-9 task to develop and test the closed loop contamination control system is to serve as a pilot plant scale intermediate test of handling fully irradiated mixed oxide fuel debris in a sodium loop system. FFTF Development Task IT-9 is intended to attack the problems of fission products in sodium systems before the problem is encountered in FFTF. The Engineering Scale Loop is an essential part of this task.

This report concludes that we cannot give assurance that the FFTF closed loops will have the ability to meet the RDT criteria of running fuel to failure, running vented and defect fuel and maintaining high availability unless the loop design and equipment and procedures are adequately proof-tested on fully irradiated defect fuel materials prior to startup. We believe that if engineering scale tests are not run now, then a trial and error learning process will be encountered on the reactor itself after the first major fuel failure, with a high probability of extensive downtime, loop modifications,

and very high costs. We believe the minimum development program responsive to the RDT approved base line conceptual description (BNWL-500) requires engineering scale tests. Only if design and procurement of the Engineering Scale Loop is started in FY-68 and construction started early in FY-69 can the development schedule remain consistent with the FFTF design and construction schedule.

FFTF DESIGN REQUIREMENTS

One of the major design requirements for FFTF is to build closed loops into a fast reactor core to allow testing fast reactor fuels up to and including fuel failure. The general design requirements for closed loop testing have been agreed upon and documented in BNWL-500⁽¹⁾ (FFTF Overall Conceptual Systems Design Description, July 7, 1967).

Significant excerpts from BNWL-500 relating to closed loop design requirements are:

Page 1-6 (K)	Plant availability target of 75%
Page 2-4 (2.2.5)	<u>Independent from Closed-Loop Operation</u> - The facility will be designed to accommodate failure of any single closed-loop fuel assembly without affecting adjacent tests and without material damage to the reactor. Meltdown within the loop is the worst consequence of failure and although loops will be designed for recovery for this evenutality, they are not intended to accommodate meltdown testing.
Page 2-7 (2.2.16)	<u>Fission Product Release Rate</u> - The plant will be designed to accommodate release of fission products from driver fuel and closed-loop test elements to the design limits of the system which will be established so as not to preclude the eventual use of vented fuel.
Page 3-25 (3.3.1)	<u>Closed Loop System</u> - (B) Requirements - Testing is to permit determination of effects and limits

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would prohibit contact maintenance, facilities will be provided to either perform remote maintenance in place, or to remotely remove and replace components.

RADIOACTIVITY IN FFTF CLOSED LOOPS

It is clear that the mission of the closed loops will result in release of significant levels of radioactivity from two sources:

- (1) Activated corrosion products from stainless steel fuel clad and structure - primarily Co^{58} (half life - 71 days) Mn^{54} (half life 314 days)
- (2) Fission products from vented fuel tests or failure of fuel cladding

This radioactivity results in high gamma radiation fields of sufficient half life to seriously complicate the operation of the system, maintenance of failed components and modifications to update equipment to meet new experimental needs.

Stainless Steel Radioisotope Transport

Assumptions

1. The activation products in 304 stainless steel were calculated using a 10^{16} nv fast flux of the FFTF spectrum.
2. The mass transfer rates are based on average corrosion rates found by GE⁽²⁾ at 10-15 ppm oxygen levels:

<u>Temperature</u>	<u>Mils/year</u>
1000°F	0.05
1100°F	0.15
1200°F	0.39
1300°F	0.95
1400°F (extrapolated)	3

The corrosion rates were averaged over the fuel pin length i.e., for 1000°F inlet and 1200°F outlet the average rate is 0.2 mils/yr. and for 1400°F outlet the average rate is 1.5 mils/yr.

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3. Mass transfer based on 217 pins 0.25 OD x 36 in. length (gas plenum surface out of the flux does not contribute significant radioactivity.)
4. Loop system - 5 MW size with IHX, pump-surge tank combination and 4-inch, schedule 10, pipe size

Total surface area	405 ft ²
Total volume	27.7 ft ³ sodium
Pump vol.	10 ft ³
Pump surface	20 ft ²
IHX vol.	5 ft ³
IHX surface	200 ft ²

5. Total radioisotope inventory in the pipe system as a function of time is approximated by

$$R = M \frac{Csat}{\lambda} (1 - e^{-\lambda t})^2$$

R = total radioisotope inventory out of reactor

M = mass transport rate/subassembly

Csat = curies of isotope at saturation in stainless steel of one subassembly

λ = decay constant

t = time

This equation computes the total curies of a given isotope in the system corrected for isotope buildup in the cladding and decay of radioisotopes already transported.

If one inserts a 217 pin subassembly into a 5 MW FFTF loop and starts a continuous irradiation (ignoring downtime) Co⁵⁸ and Mn⁵⁴ rapidly build up in the cladding. These radioisotopes move by mass transfer out of the core where decay continues. The result is an "S" shaped curve in Fig. 1 and 2 giving the buildup of total curies of Co⁵⁸ and Mn⁵⁴. The Co⁵⁸ will reach a steady state in the system in about 300 days and the Mn⁵⁴ will continue to accumulate for about five years reaching a total saturated value of 450 Ci in the system at 1200F max. clad temp. and 3000 Ci in the system for a max. clad temperature of 1400F. These values represent upper limits since some decay will occur at each reactor refueling period, and corrosion rates could be

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FIGURE 1. Buildup of Activity in 5 MW FFTF Closed Loop Maximum Fuel Clad 1400 °F

FIGURE 2. Buildup of Activity in 5 MW FFTF Closed Loop Maximum Fuel Clad 1200 °F

substantially lower if the oxygen levels were lower. In many cases less than 217 pins will be tested which will reduce the inventory of radioisotopes proportionately.

Assuming continuous operation for one year, typical decay curves are given in Figures 3 and 4.

Failed Fuel Fission Product Release

Assumptions

1. Fission product inventory based on buildup in FFTF driver fuel.
2. 60% of the volatile fission gas formed is released to the fuel pin gas plenum.
3. A pin hole failure releases that fraction of the gas to equalize internal and external pressures.
4. Loop system - same as for radioisotope transport calculation.
5. Na^{24} inventory - saturated; Na^{22} inventory-3 yr. irradiation.

Four closed loop fuel failure accidents were considered and the curies of mixed fission products calculated as a function of days burnup prior to failure. These cases are tabulated in Table 1 for:

Release of volatile fission products only through pin hole
 Contents of one pin released
 " " one subassembly (217) pins

Decay curves after fuel failure (and shutdown) are given in Figures 5-8. When the decay of fission products is compared with the Na^{24} and Na^{22} inventories, it is interesting to note in Figure 3 that the fission products released from a single pin with 23 days irradiation is about 4×10^4 curies and is initially about the same order as the Na^{24} activity in the loop.

Vented Fuel

The amount of fission products released during vented fuel operation is very strongly dependent on the characteristics of the vent. If all the volatile fission products which escaped from the fuel (60%) are released to the sodium the release rate would be about 2×10^5 Ci/day per 217 pin element. However, one GE vented fuel test (3) released only fractions of 10^{-8} to 10^{-6}

FIGURE 3. Decay of Activity in 5 MW Closed Loop After One Year of Continuous Operation with One 217 Pin Fuel element with Maximum Fuel Clad
 $T = 1400^{\circ}\text{F}$

FIGURE 4. Decay of Activity in 5 MW Closed Loop After One Year Continuous Operation with One 217 Pin Fuel Element with Maximum Fuel Clad $T = 1200^{\circ}\text{F}$

TABLE I
FISSION PRODUCT RELEASE FROM FFTF FUEL FAILURES

		Curies/Pin Available as Gas						
		Decay						
		Shutdown	1 Minute	1 Hour	1 Day	10 Days	100 Days	1 Year
11.57d Burnup	Noble Gas	6.8×10^3	4.0×10^3	2.0×10^3	9.4×10^2	2.0×10^2	6.8×10^{-2}	4.7×10^{-2}
	Halogens	5.3×10^3	4.0×10^3	3.0×10^3	1.1×10^3	2.0×10^2	5.4×10^{-2}	3.3×10^{-3}
	Vol.Solids	7.2×10^3	5.0×10^3	2.0×10^3	6.5×10^2	1.2×10^2	6.6×10^0	9.0×10^{-1}
23.15d Burnup	Noble Gas	7.2×10^3	4.2×10^3	2.3×10^3	1.1×10^3	2.6×10^2	1.3×10^{-1}	9.6×10^{-2}
	Halogens	5.8×10^3	3.9×10^3	3.2×10^3	1.3×10^3	2.6×10^2	7.8×10^{-2}	6.6×10^{-7}
	Vol.Solids	7.4×10^3	5.0×10^3	2.2×10^3	7.2×10^2	1.6×10^2	1.3×10^1	1.7×10^0
115.7d Burnup	Noble Gas	7.2×10^3	4.2×10^3	2.4×10^3	1.2×10^3	2.8×10^2	5.4×10^{-1}	4.7×10^{-1}
	Halogens	5.8×10^3	4.0×10^3	3.3×10^3	1.3×10^3	2.8×10^2	8.4×10^{-2}	3.3×10^{-6}
	Vol.Solids	7.8×10^3	5.0×10^3	2.2×10^3	8.4×10^2	2.8×10^2	4.0×10^1	9.6×10^1
1157d Burnup	Noble Gas	8.4×10^3	5.8×10^3	3.3×10^3	1.7×10^3	4.3×10^2	5.4×10^0	5.0×10^0
	Halogens	5.8×10^3	4.3×10^3	3.5×10^3	1.4×10^3	3.2×10^2	1.0×10^{-1}	2.3×10^{-6}
	Vol.Solids	9.0×10^3	6.0×10^3	3.1×10^3	1.5×10^3	7.2×10^2	2.2×10^2	1.5×10^2

Curies Fission Products/Pin (Available)

		Decay						
		Shutdown	1 Minute	1 Hour	1 Day	10 Days	100 Days	1 Year
11.57d Burnup	Noble Gas	1.1×10^4	6.6×10^3	3.4×10^3	1.6×10^3	3.4×10^2	1.1×10^{-1}	7.8×10^{-2}
	Halogens	8.9×10^3	6.6×10^3	5.0×10^3	1.8×10^3	3.4×10^2	9.0×10^{-2}	5.5×10^{-7}
	Vol.Solids	1.2×10^4	8.4×10^3	3.4×10^3	1.1×10^3	2.0×10^2	1.1×10^1	1.5×10^0
	All Other F.P.'s	6.6×10^4	5.0×10^4	2.2×10^4	8.4×10^3	2.3×10^3	3.4×10^2	5.3×10^1
23.15d Burnup	Noble Gas	1.2×10^4	7×10^3	3.8×10^3	1.4×10^3	4.3×10^2	2.1×10^{-1}	1.6×10^{-1}
	Halogens	9.6×10^3	6.5×10^3	5.3×10^3	2.1×10^3	4.3×10^2	1.3×10^{-1}	1.1×10^{-6}
	Vol. Solids	1.3×10^4	8.4×10^3	3.6×10^3	1.2×10^3	2.6×10^2	2.1×10^1	2.9×10^0
	All Other F.P.'s	6.6×10^4	5.2×10^4	2.3×10^4	1.0×10^4	3.6×10^3	6.0×10^2	1.0×10^2

Curies Fission Products/Pin (Available)

(Continued from previous page)

Decay

		Shutdown	1 Minute	1 Hour	1 Day	10 Days	100 Days	1 Year
115.7d Burnup	Noble Gas	1.2×10^4	7.0×10^3	4.0×10^3	2.0×10^3	4.6×10^2	9.0×10^{-1}	7.8×10^{-1}
	Halogens	9.6×10^3	6.6×10^3	5.5×10^3	2.2×10^3	4.6×10^2	1.4×10^{-1}	5.5×10^{-6}
	Vol.Solids	1.3×10^4	8.4×10^3	3.7×10^3	1.4×10^3	4.6×10^2	6.6×10^1	1.6×10^1
	All Other F.P.'s	7.2×10^4	5.8×10^4	3.0×10^4	1.6×10^4	7.8×10^3	2.1×10^3	4.4×10^2
1157d Burnup	Noble Gas	1.4×10^4	9.6×10^3	5.5×10^3	2.9×10^3	7.2×10^2	9.0×10^0	8.4×10^0
	Halogens	9.6×10^3	7.2×10^3	5.8×10^3	2.4×10^3	5.3×10^2	1.7×10^{-1}	4.8×10^{-6}
	Vol.Solids	1.5×10^4	1.0×10^4	5.1×10^3	2.5×10^3	1.2×10^3	3.6×10^2	2.5×10^2
	All Other F.P.'s	8.4×10^4	7.2×10^4	4.3×10^4	2.6×10^4	1.4×10^4	5.0×10^3	1.7×10^3

Curies Fission Products Available/Subassembly

Decay

		Shutdown	1 Minute	1 Hour	1 Day	10 Days	100 Days	1 Year
11.57d Burnup	Noble Gas	2.5×10^6	1.4×10^6	7.4×10^5	3.4×10^5	7.4×10^4	2.5×10^1	1.7×10^1
	Halogens	1.9×10^6	1.4×10^6	1.1×10^6	3.9×10^5	7.4×10^4	2.0×10^1	1.2×10^{-4}
	Vol.Solids	2.6×10^6	1.8×10^6	7.4×10^5	2.4×10^5	4.4×10^4	2.4×10^3	3.3×10^2
	All Other F.P.'s	1.4×10^7	1.1×10^7	4.8×10^6	1.8×10^6	5.0×10^5	7.4×10^4	1.2×10^4
23.15d Burnup	Noble Gas	2.6×10^6	1.5×10^6	8.3×10^5	4.2×10^5	9.4×10^4	4.6×10^1	3.5×10^1
	Halogens	2.1×10^6	1.4×10^6	1.2×10^6	4.6×10^5	9.4×10^4	2.8×10^1	2.4×10^{-4}
	Vol.Solids	2.8×10^6	1.8×10^6	7.9×10^5	2.6×10^5	5.7×10^4	4.6×10^3	6.4×10^2
	All Other F.P.'s	1.4×10^7	1.1×10^7	5.0×10^6	2.2×10^6	7.9×10^5	1.3×10^5	2.2×10^4
115.7d Burnup	Noble Gas	2.6×10^6	1.5×10^6	8.8×10^5	4.4×10^5	1.0×10^5	2.0×10^2	1.7×10^2
	Halogens	2.1×10^6	1.4×10^6	1.2×10^6	4.8×10^5	1.0×10^5	3.1×10^1	1.2×10^{-3}
	Vol.Solids	2.8×10^6	1.8×10^6	8.1×10^5	3.1×10^5	1.0×10^5	1.4×10^4	3.5×10^3
	All Other F.P.'s	1.6×10^7	1.3×10^7	6.6×10^6	3.5×10^6	1.7×10^6	4.6×10^5	9.6×10^4
1157d Burnup	Noble Gas	3.1×10^6	2.1×10^6	1.2×10^6	6.4×10^5	1.6×10^5	2.0×10^3	1.8×10^2
	Halogens	2.1×10^7	1.6×10^6	1.3×10^6	5.3×10^5	1.2×10^5	3.7×10^1	1.1×10^{-3}
	Vol.Solids	3.3×10^6	2.2×10^6	1.1×10^6	5.5×10^5	2.6×10^5	7.9×10^4	5.5×10^4
	All Other F.P.'s	1.8×10^7	1.6×10^7	9.4×10^6	5.7×10^6	3.1×10^6	1.1×10^6	3.7×10^5

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FIGURE 5. Activity Inventory in a FFTF 5 MW Closed Loop After Release of 100% of the Fission Products of One Fuel Pin Irradiated for 23 Days.

FIGURE 6. Activity Inventory in FFTF 5 MW Closed Loop After Release of 100% of the Fission Products of One Fuel Pin Irradiated for 1157 Days

FIGURE 7. Activity Inventory in a FFTF 5 MW Loop After Release of 100% of the Fission Products in a FFTF Type Subassembly After 23 Days of Irradiation

FIGURE 8. Activity Inventory in a FFTF 5 MW Loop After Release of 100% of the Fission Products in a FFTF Type Subassembly After 1157 Days of Irradiation

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of the fission products so the release rate of 217 pins would be about 0.2 Ci/day using a release fraction of 10^{-6} . The calculation of the steady state content in the loop has not been done because of the very complicated number of half lives and decay schemes requires a computer analysis be programmed. However, WP Kunkel⁽⁴⁾ has roughly estimated the FFTF reactor would have about 5 curies of fission products in the sodium if operated with vented fuel. Adjusting this number downward to one subassembly and correcting for a different sodium volume in the closed loop vs. the reactor gives a crude estimate of 1 Ci at steady state in the closed loop if a 217 pin vented fuel element were tested. It appears the vented fuel release is significantly less important than either the corrosion product or failed fuel problem.

Radiation Levels in Closed Loop Cell

For purposes of illustration an estimate has been made of the radiation levels in an FFTF equipment cell assuming that equipment is placed in the room and no local shielding of components provided. Since the activated corrosion products build up over a long time we assumed one year of operation before a mild fuel failure of only one pin with 23 day irradiation. Even for this small fission product release it will be seen that the fission product radiation is orders of magnitude greater than the activated corrosion products.

The following procedure was used to estimate the radiation level in an FFTF closed loop cell. Operating conditions were assumed to be 1 year of reactor operation at 1200°F maximum fuel cladding temperature at which time a new subassembly was inserted at that time. 23.15 days later, one fuel pin in the new subassembly failed. The inventory of the radioactive species in one loop under these conditions is the following:

4×10^4 curies of fission products
 4×10^4 curies of Na^{24} (half-life 15h)
173 curies of Co^{58} (half-life 71 days)
76 curies of Mn^{54} (half-life 313 days)

The last two values were taken from Fig. 1 and 2 of activation product radiation buildup in the FFTF piping. All the fission products were assumed to mix evenly throughout the sodium volume. All the cobalt and manganese was assumed to deposit uniformly on the piping surface. Uniform fission product mixing is probably reasonable in the first stages after

a fuel failure, however, a large fraction of the fission products will "plate out" on the piping in the first few weeks. The status of information on this aspect is covered in the next section.

A simplified model of an FFTF closed loop cell was assumed, as shown in Fig. 9. Descriptions of the parts are given in Table 2. From these data, calculations were made by C. Mansius of PNL, using computer programs ISOHLD and RIBD to obtain the radiation level (R/hr) inside the cell. This program corrects for each fission product energy and decay scheme as a function of time as well as correcting for geometry and self shielding effects.

The assumed radioisotope distribution is given in Table 3 and the assumed geometry given in Fig. 9. The R/hr. dose rate from each source is tabulated in Table 4 and the decay curves plotted in Fig. 10. In this example the dose rate from the fission products is initially about 100 times higher than the corrosion products and is still 10 times higher than the corrosion products after 6 weeks of cooling. Obviously, if the fuel had been in the reactor longer than the assumed 23 days before the failure, the long lived fission products would predominate and the fission product dose rate at 1000 hours would be higher. If a 1400F fuel clad temperature were assumed to increase Co^{58} and Mn^{54} activity by a factor of 10, the fission products would still produce more radiation for about a month after failure.

One can conclude that except for very small fuel "leakers," fission product radiation will be higher than corrosion product radiation. If a major fuel failure occurs the fission product radiation will predominate by several orders of magnitude.

One very interesting result of the calculation is that much of the radiation in this assumed case comes from the pipes rather than the IHX or pump (see Table 5). This is partly due to the fact that measuring point X is closer to the pipes, but another major factor is a considerable self-shielding effect caused by the large mass of stainless and sodium in the large components.

This computer calculation can calculate radiation levels with various cases of assumed shielding around components. All that is needed is valid data on the distribution of radioisotopes in the piping system. The values used in Fig. 1 are just assumptions; radiation deposition patterns based

FIGURE 9. Model of FFTF Cell (Not to scale)

FIGURE 10. Cell Dose Rate in Closed Loop Cell Function of Decay Time

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TABLE 2
DESCRIPTION OF PARTS IN FFTF LOOP CELL MODEL

<u>Part</u>	<u>Description</u>	<u>Area*</u>	<u>Volume*</u>
A, B	Stainless pipe, 4" i.d., 0.120" wall, 20' long	21 ft ²	175 ft ³
C, D	Stainless pipe 4" i.d., 0.120" wall, 3' long	3.15 ft ²	0.225 ft ³
Pump	Rt. circular cylinder, 10' high x 18" dia. Casing 0.120" thick. 47% of volume is stainless steel	20 ft ²	10 ft ³
IHX	Rt. circular cylinder 10' high x 18' dia. Case 0.120" thick contains 121 10' long 1/2" schedule 40 SS tubes (0.840" o.d., 0.622" i.d.) 25% of volume is stainless steel	200 ft ²	5 ft ²
	Entire loop, inside and outside cell	405 ft ²	27.7 ft ³

* Area contacted by primary sodium, volume of primary sodium

TABLE 3
ASSUMED DISTRIBUTION OF RADIOISOTOPES - CASE I

<u>Component</u>	<u>Fission Products</u>	<u>²⁴Na</u>	<u>⁵⁸Co</u>	<u>⁵⁴Mn</u>
Pipes A + B	5040 Ci	5040 Ci	18 Ci	7.9 Ci
Pipes C + D	780	750	2.7	1.18
Pump E	14,400	14,400	8.56	3.76
IHX F	7,200	7,200	85.6	37.6

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TABLE 4
CALCULATED DOSE RATES - CASE I
DOSAGE AT X

	<u>Pipes</u> <u>A + B</u>	<u>Pipes</u> <u>C + D</u>	<u>Pump</u> <u>E</u>	<u>IHX</u> <u>F</u>	<u>Total</u>
Cobalt 58 (t = 0)	1.8 R/hr	.13 R/hr	.09 R/hr	.94 R/hr	2.96 R/hr
Manganese 54 (t = 0)	.77	.05	.04	.41	1.27
Sodium 24 (1 hr)*	1900	116	760	411	3190
Fission Products (1 hr)*	264	16.5	82.8	45.5	409
Fission Products (10 hr)*	127	8.0	36.8	20.3	192
Fission Products (100 hr)*	50.9	3.2	15.0	8.2	77.3
Sodium 24 (10 hr)*	1200	73	478	259	2010
Sodium 24 (100 hr)*	19	11.6	7.6	.4	39
Sodium 24 (1000 hr)*					
FP's (1000 hr)*					= 10/hr

* decay times

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on measurements of isotope distribution in a prototypical loop as proposed in Task IT-9 are required to reliably predict radiation level in the cell. Some additional examples of cell radiation levels with and without local shielding are given in the appendix. In Case II which assumes 99% of all activated corrosion products deposit in the IHX (which is shielded), the radiation levels are only 30-40 mr/hr after 1 year of operation. However, if a fission product release of 100% of 1 pin occurs cell radiation levels increase 1000 fold.

RADIOISOTOPE DEPOSITION PATTERNS

Present knowledge does not allow reliable prediction of the specific locations where radioactivity will deposit in the FFTF System since none of the mechanisms are understood. Hopefully these mechanisms will eventually be established and current laboratory programs are being oriented to that goal. It is expected that predictions based on results of irradiated fuel release tests in the IT-9 Engineering Scale Loop will be far more accurate than any speculations that can be made today. However, one can review available knowledge on deposition, make some assumptions and calculate some examples.

Activated Corrosion Products

Transport of corrosion product radioactivity has not been a significant problem in sodium reactors to date. However, the high flux of FFTF increases the activation products in the stainless steel fuel cladding about a factor of 100 higher than a reactor such as SRE. The proposed operating temperatures of 1200F-1400F will increase corrosion rates at least a factor of 10 higher than SRE leading to the estimate that the radioisotope transport problem in FFTF will be at least a factor of 1000 worse than has been previously encountered.

Deposited corrosion product radioactivity was observed on piping near the cold trap in SRE⁽⁵⁾ at levels of about $0.02 \mu\text{Ci}/\text{cm}^2$. In LAMPRE sodium contained some Cr^{51} , Co^{58} and Mn^{54} but levels were not identified⁽⁶⁾. Activation of corrosion products at Hallam and FERMI were not significant.

Little information is available on deposition processes of Co^{58} and Mn^{54} in sodium loops since most existing data center about the hot leg weight loss measurements. However, some studies have been conducted in which deposition coupons were placed in the colder parts of the system and weight increases observed. In general, the drop in temperature in the heat exchanger causes

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supersaturation of the iron and nickel and deposition occurs. But since the activated corrosion products represent a very small fraction of the mass transferred there can be no assurance that deposition of the radioisotopes will occur. Even for the iron and nickel significant deposits are sometimes found downstream of the heat exchanger indicating time dependent deposition mechanisms.

Fission Product Deposition

Past experience has shown major fission product release in a sodium reactor is very costly and cladding failures involving domestic reactors have resulted in downtimes of about a year. It would be difficult to differentiate the time loss due to decay of fission products and induced activity from the time loss for repairs and modifications which inevitably follow a reactor failure. Most of the lost time is taken up with the dismantling, cleanup, and study of the failure, but the radiation fields encountered undoubtedly have a major impact on the time and manpower to accomplish repairs.

The fission products can be divided into chemical groups: halogens, noble gases, volatile solids and non-volatiles. The halogens react rapidly with the sodium coolant, generally remain suspended (subject to coolant and wall parameters, hot and cold trapping, and containment geometry). The noble gases rapidly disengage from the coolant and become a part of the inert plenum gas. The "bad actor" of the fission noble gases is Xe^{137} (2.3 day half life) which decays to long lived (30 yr. half life) Cs^{137} which in turn attaches to the inaccessible plenum interior surfaces. The volatile solids (Cs, Te, Se) distribute throughout the piping and gas systems. Tellurium and selenium are expected to form tellurides and selenides although information on vapor pressure of the elements (especially tellurium) as a function of temperature, rate of formation, solubility, "trapability" and "suspendability" of the compounds is quite meager. Cesium is expected to be divided between coolant and cover gas subject to temperature, and equilibrium time. This cesium will add to that resulting from xenon decay. Some of the cesium in the sodium is expected to "plate out" to an extent dependent on coolant and wall conditions, trapping, and containment geometry.

Examination of the sodium after fuel failures at LAMPRE, SRE, BR-5 and FERMI all showed that major fractions of the fission products were deposited on the pipe walls within a few weeks after the fuel failure. Examination ⁽⁷⁾ of SRE

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after the fuel failure showed the fission products had diffused into the pipe walls about 0.1 mil. The difficulties experienced by the Russians⁽⁸⁾ in removing radioisotopes from BR-5 piping also indicates direct alloying with the stainless steel. Diffusion of the fission products into the pipe walls makes contamination in sodium reactors inherently different from water reactors where contamination is incorporated into oxide films on the metal surface. Thus, decontamination of aqueous reactors utilizes solutions to remove oxide films whereas decontamination of sodium systems will require controlled dissolution of the stainless steel.

Fast reactor incidents to date have not presented a contamination problem of the magnitude of possible fuel failures in an FFTF closed loop. As seen from Table 5 the expected activity from the meltdown of one FFTF type subassembly will release fission products of three to four orders of magnitude more than any domestic sodium reactor accident to date. Even a 1% release of the fission products in a subassembly will be a factor of 10 higher than the SRE accident.

Three fuel cladding failures in the LAMPRE contributed low fission product contamination because of low burnup. The released plutonium (from about 160g Pu + Fe eutectic) generally deposited on the piping (forming a Pu-Fe eutectic); the coolant showed < 1 ppm Pu.

The FERMI meltdown presumably occurred from partial blockage of the coolant. The nature of the blockage has not been published as yet. In any event, the contamination problem resulting from the release of fission products was minor because of the low burnup ($\sim 0.06\%$). An interesting comparison is the FFTF will achieve in 2 days at 400 MW the same burnup FERMI had when failure occurred.

The SRE failed due to partial coolant blockage from deposition of carbonaceous material which resulted from tetralin leakage into the primary sodium and subsequent decomposition. This carbonaceous material demonstrated the ability to absorb 10^3 to 10^4 times as much fission products as the sodium. Only Kr and Xe were identified in the plenum gas above the 960°F sodium. A very useful radiation survey of SRE was made prior to demolition.⁽⁹⁾ This survey (Fig. 11) indicated the hottest region was the cold leg isothermal outlet pipe of the heat exchanger with significant activity also accumulating in and around the pump. These radiation levels indicate that deposition is not directly correlated with temperature.

FIGURE 11. SRE* Main Gallery Radiation Levels
55 to 61 Days After Shutdown

* SRE and HNPF Opening and Modification Experience, ANS Fast Reactors Topical Meeting, April 10-12, 1967, San Francisco.

TABLE 5 - SUMMARY OF SODIUM REACTOR FUEL FAILURES

Reactor	Power at Failure MW	Failed Fuel Elements/ Total Fuel Elements	Approximate FP Inventory Released to Coolant, Ci	Reactor Down Time
LAMPRE	1	3/140	2.3	--
FERMI	34	2/105	$\sim 6 \times 10^3$	12 mo.
SRE	20	13/43	$\sim 10^4$	14 mo.
BR-5	5	18/81	$> 3 \times 10^5$ ***	6 mo.
FFTF	400	1/76*	$\sim 10^7$ **	--
FFTF closed loop	5	1/1*	$\sim 10^7$ **	--

* assumed

** one subassembly @ goal expense

*** best estimate from available data

Information is incomplete on the Russian BR-5 incident so best estimates are calculated from available literature. The cladding failure was apparently due to fuel swelling and clad embrittlement. The reactor operated at various power levels and outlet coolant temperature with failed cladding on some of the fuel element for about a year. After shutdown and two weeks cooling, the gamma background in the primary coolant was "several thousand micro-R per second" (several R/hr) at an unspecified range⁽¹⁰⁾. Transfer of the coolant to storage did not appreciably decrease the radiation level. Steam treatment decreased the activity to one-half of the original. 5% nitric acid at 70°C for 6 hours reduced the activity a factor of 1.5 while two additional treatments further reduced the activity two-fold. Then five cycles of flushes, consisting of (1) 0.5% KMnO_4 at 70°C for 24 hours followed by (2) 5% HNO_3 and 1% oxalic acid at 70°C for 3-4 hours following by (3) rinsing with distilled water resulted in a background of 10 uR/sec at this point.⁽¹¹⁾ This indicated a decontamination factor of better than 200 was achieved. The core was reloaded and operation restarted in March 1962. Apparently BR-5 operated at least to March 1964 at which time the PuO_2 elements achieved 5.8% burnup. No additional

information has been forthcoming. This experience demonstrates that a sodium reactor can be decontaminated successfully and restarted.

Fission Product Removal

About 2% of the curies released in the SRE core accident were collected in the cold trap⁽¹²⁾, however, the presence of a large carbon particle content complicates interpretation, since carbon readily absorbs fission products and the cold trap could have filtered out the carbon particles. LASL data⁽¹³⁾ shows clearly the ability of carbon to remove fission products and also that cold traps can remove some fission products. Data to calculate removal of mixed fission products from an FFTF loop are not yet available.

One major goal of FFTF Task IT-9 is testing of promising fission product control systems possibly including:

1. Mechanical removal of fuel debris
 - (a) filter
 - (b) hydroclone
2. Fission product traps
 - (a) cold trap
 - (b) two cold traps in series first add large oxygen contents to be removed in the second trap and co-precipitate the fission products
 - (c) traps with activated charcoal
 - (d) traps with activated alumina (pure Al_2O_3 is stable in sodium)
3. Chemical equilibrium adjustment - for example Cs^{137} distributes between sodium and gas

$$Cs^{137} \rightleftharpoons Cs_{gas}^{137}$$

If we add chemical cesium so the ratio of nonactive to active atoms is say

$$\frac{Cs}{Cs^{137}} = 10^9$$

then one would expect to reduce the active Cs^{137} atoms in the gas by a factor of 10^9 . The same should be true of diffusion into stainless steel. By keeping the Cs^{137} in solution it can be drained out of the system with the sodium.

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4. Temperature, flow and residence time adjustment. By cooling the sodium it may be possible to selectively precipitate some fission products.

FFTF DESIGN PROBLEMAlternatives

The FFTF designer must find solutions to accommodate the released radioactivity, provide assurance that the desired fuel tests can be run, that the system can be kept operational, and the system can be replaced when desired. Possible design solutions include:

Alternative 1 - Use only 100% replaceable package loops - The numerous studies leading to FFTF design requirements have clearly shown the package loops will not meet all testing needs and loops with components external to the reactor are required. (BNWL-500 page 1-6).

Alternative 2 - Remote Operated Canyon - Loop equipment would be installed in a cell with remote manipulators and remote tools for repair and modification. This solution makes each loop a hot cell with very high costs. Numerous remote tools must be developed since all maintenance would be done in the cell.

Alternative 3 - Loop Modules Designed for Remote Replacement - This solution would provide tools to remotely remove and replace a limited number of modules, but no maintenance would be done on a module inside the cell. Economics dictates that a decontamination cell and repair facilities be provided for module repair.

Alternative 4 - Build redundant highly reliable system - Design a system that will never need repair. This is probably impractical and costs would be prohibitive.

The best solution appears to be Alternative 3 to provide for remotely replaceable modules since:

1. Repairs can be done under some controlled conditions
2. Module replacement gets the reactor back in operation quickly
3. New equipment can be built and tested in a module and reactor down time minimized for installation.

Once the designer has chosen the module concept he must decide the following:

1. If a module must be replace for every minor failure of a wire, heater, thermocouple or valve, costs and downtime will still be high. Will it be possible to shield radioactive components so that contact maintenance will be possible most of the time on the rest of the system?
2. Which components will be radioactive? What components go into what module?
3. How much circulating fission products and fissionable fuel can be tolerated in the sodium?
4. What methods or traps will be installed to remove fission products and fuel rupture debris from the sodium?
5. On what basis can the chosen fission product control system be proven effective?
6. Do we want to test the remote installation and replacement of loop modules prior to FFTF startup?
7. What methods and tools will be needed to replace a module?
8. How is a module to be decontaminated and repaired?
9. On what basis will the system operating procedure and limits for handling fuel failures be chosen?

Closed Loop Operation with Extensive Fuel Failure Debris in the Loop

If it is assumed the closed loop could be built tightly enough to contain the fission products and if any degree of remote maintenance could be accomplished, the problems created for continued use of a grossly contaminated closed loop facility for further tests of new fuel should be considered:

1. No subsequent fuel test would have fuel failure monitoring since the fuel circulating through the core would release both fission gas and delayed neutrons. Will safety considerations allow operation without fuel failure detection?
2. No vented fuel tests could be run where fission product release was to be measured.
3. It would be difficult to analyze the mechanism of a fuel failure if the loop is already full of debris from a previous fuel failure.

4. Fuel assemblies and components removed would have gross surface contamination.
5. Surface deposits of fissionable fuel could affect heat generation or corrosion of test elements.
6. Fuel debris represents a significant chemical contamination of the sodium and no future test would have prototypical coolant chemistry.
7. Potential plugging could occur in future tests by sudden release of deposits.

It would appear that even if operation with a grossly contaminated loop were technically feasible the disturbances caused to future experiments would be unacceptable.

DEVELOPMENT PLAN

The present plan to obtain information and test the design adequacy of the system incorporates tracer level experiments at AI, LASL, BNL and PNL (Tasks RS-8 and RS-9). We hope laboratory scale studies will provide sufficient data for the prediction of deposition patterns of activated corrosion products. These laboratory studies are also essential to understanding necessary to predict what will happen after a fuel failure. However, the laboratory studies can never really come to grips with the engineering scale problems of fuel failure because fully irradiated mixed oxide has different characteristics from tracer level irradiated fuel. Since many of the rupture debris deposition patterns will be flow or geometry sensitive, it is dangerous to extrapolate from bench scale tests to full reactor scale. The IT-9 task to develop and test the closed loop contamination control system is to serve as a pilot plant scale intermediate test of handling fully irradiated mixed oxide fuel debris in a sodium loop system.

We cannot give assurance that the FFTF closed loops will have the ability to meet the RDT criteria of running fuel to failure, running vented and defect fuel and maintaining high availability unless the loop design and equipment and procedures are proof tested on fully irradiated defect fuel materials prior to startup. We believe that if engineering scale tests are

not run now, then a trial and error learning process will be encountered on the reactor itself after the first major fuel failure, with a high probability of extensive downtime, loop modifications, and very high costs. We believe the minimum development program responsive to the RDT approved base line conceptual description (BNWL-500) requires engineering scale tests. Only if design and procurement of the Engineering Scale Loop is started in FY-68 and construction started early in FY-69 can the development schedule remain consistent with the FFTF design and construction schedule.

Specific items the Engineering Scale Loop is expected to accomplish are:

1. Define the loop piping module layout and component orientation to optimize operation and maintenance after a fuel failure.
2. Develop and proof test a fission product control system that can be demonstrated to work satisfactorily.
3. Establish the locations where radioactivity will deposit so that deposition areas can be eliminated, isolated or shielded.
4. Define the proper loop operation and shutdown procedures after a fuel failure to allow the closed loop operating manual to be based on demonstrated performance rather than best guesses.
5. Continue fission product containment tests of nozzle caps, seals, valves, gas samplers, etc., and perform realistic tests of fission gas detectors and sodium sampler for fission product analysis.
6. Develop and test remote operation and maintenance procedures directly applicable to the FFTF closed loop.
7. Test remote maintenance methods. The remote operation and maintenance of this loop will allow actual training to begin of PNL craftsmen and foremen on the problems and hazards of maintenance of highly radioactive sodium loop systems. This nucleus of training effort for about three years prior to FFTF startup will be of considerable value in assuring an adequately trained maintenance force for FFTF.

8. Results of tests on amount of radioactive rupture debris that can be removed by conventional techniques of draining and cold trap circulation will clearly define the need for aqueous decontamination procedures. Since any aqueous process will never be rated as safe for reactor use until test loop runs are successful, the Engineering Scale Loop will be needed if FFTF is ever to have a decontamination method.

It is premature to suggest the specific tests to be run in the Engineering Scale Loop since about two years of laboratory scale screening studies are to be done first. However, the FFTF schedule does not permit all decisions to be based on information developed in sequence and a decision must be made now if we are to run engineering scale contaminated systems tests three years from now.

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APPENDIX

Radiation dose levels in an FFTF closed loop cell for three additional deposition patterns were also calculated by C. Mansius, using the same reactor operating conditions used in Case I discussed in the text. (1 year operation, insertion of new fuel subassembly, failure of a single fuel pin after 23.15 days operation with 100% release of fission products in the pin into the sodium.) The closed loop cell geometry is as shown in Figure 9. The deposition and shielding schemes are these:

- Case I See text page 19.
- Case II Uniform deposition of fission products, ^{58}Co , and ^{54}Mn , on stainless steel surfaces.
- Case III Uniform deposition of fission products on the stainless steel surfaces, 99% of ^{58}Co and ^{54}Mn deposited on intermediate heat exchanger (IHX) surfaces. Lead shield 12" thick around IHX. This might be a typical case if corrosion products plate out on the temperature drop in the IHX, but fission products are spread throughout the system.
- Case IV 99% of fission products, ^{58}Co , and ^{54}Mn deposited on IHX surfaces. Lead shield 12" thick around IHX.

The results are given in Table A-1. When the IHX is shielded with 12" of lead, the dose rate from activity deposited on the IHX surfaces is less than 0.001 R/hr at all times.

The assumption that fission product distribution is uniform with sodium volume (Case I) vs. uniform with SS area (Case II) makes little difference in cell radiation levels. If the deposition locations for ^{58}Co and ^{54}Mn are effectively shielded as in Case III, cell radiation levels due to corrosion products are reduced by a factor of 100. In Case IV by assuming 99% of the activity can be trapped in a single, well shielded location, cell radiation levels are reduced by a factor of 80 after 100 hours of decay (Case IV total at 100 hr. of 0.93 R/hr. vs. Case II total at 100 hr. of 75 R/hr.).

TABLE A-1
RADIATION DOSE RATE IN CENTER OF CLOSED LOOP CELL

<u>Case II</u> <u>Source Mat'l.</u>	<u>Decay Time</u> <u>Hours</u>	<u>Source Position (see Fig. 9)</u>						<u>Total</u> <u>(R/hr)</u>
		<u>A</u>	<u>B</u>	<u>C</u>	<u>D</u>	<u>E</u>	<u>F</u>	
		<u>20 ft. pipes</u>		<u>3 ft. pipes</u>		<u>Pump</u>	<u>IHX</u>	
²⁴ Na	1	950	950	58	58	760	411	3190
²⁴ Na	10	600	600	36.5	36.5	478	259	2010
²⁴ Na	100	9.5	9.5	0.6	0.6	7.6	4.1	32
F.P.	1	109	109	6.4	6.4	11.2	12.2	364
F.P.	10	52.4	52.4	3.1	3.1	4.9	54.2	170
F.P.	100	21.0	21.0	1.2	1.2	2.0	2.0	68.4
⁵⁸ Co	1-100	0.9	0.9	0.065	0.065	0.09	0.94	2.96
⁵⁴ Mn	1-100	0.385	0.385	0.025	0.025	0.04	0.41	1.27

TABLE A-1

RADIATION DOSE RATE IN CENTER OF CLOSED LOOP CELL
R/hr.

Case III

<u>Source Mat'l</u>	<u>Decay Time Hours</u>	<u>Source Position</u>						<u>Total (R/hr)</u>
		A <u>20 ft. pipes</u>	B	C <u>3 ft. pipes</u>	D	E <u>Pump</u>	F* <u>IHX</u>	
²⁴ Na	1	950	950	58	58	760	0.001	2778
²⁴ Na	10	600	600	36.5	36.5	478	0.001	1751
²⁴ Na	100	9.5	9.5	0.6	0.6	7.6	0.001	27.9
F.P.	1	109	109	6.4	6.4	11.1	0.001	242
F.P.	10	52.4	52.4	3.1	3.1	4.9	0.001	115.6
F.P.	100	21.0	21.0	1.2	1.2	2.0	0.001	46
⁵⁸ Co	1-100	0.02	0.02	0.001	0.001	0.001	0.001	0.044
⁵⁴ Mn	1-100	0.01	0.01	0.0003	0.0003	0.0005	0.001	0.022

* Shielded with 12 inches of lead

TABLE A-1
RADIATION DOSE RATE IN CENTER OF CLOSED LOOP CELL
R/hr

<u>Case IV</u>								
<u>Source Mat'l</u>	<u>Decay Time Hours</u>	<u>Source Position</u>		<u>Source Position</u>		<u>E</u>	<u>F*</u>	<u>Total</u>
		<u>A</u>	<u>B</u>	<u>C</u>	<u>D</u>	<u>Pump</u>	<u>IHX</u>	<u>(R/hr)</u>
		<u>20 ft. pipes</u>		<u>3 ft. pipes</u>				
^{24}Na	1	950	950	58	58	760	0.001	2779
^{24}Na	10	600	600	36.5	36.5	478	0.001	1751
^{24}Na	100	9.5	9.5	0.6	0.6	7.6	0.001	27.9
F.P.	1	2.2	2.2	0.054	0.054	0.22	0.001	4.73
F.P.	10	1.0	1.0	0.026	0.026	0.10	0.001	2.15
F.P.	100	0.4	0.4	0.01	0.01	0.04	0.001	0.861
^{58}Co	1-100	0.02	0.02	0.001	0.001	0.001	0.001	0.044
^{54}Mn	1-100	0.01	0.01	0.0003	0.0003	0.0005	0.001	0.022

* Shielded with 12 inches of lead

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