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RADIOACTIVE MASS TRANSPORT
IN FFTF PRIMARY PIPING

W. F. Brehm

August 1968

AEC RESEARCH &
DEVELOPMENT REPORT

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January 15, 1969

BNW/FFTF-69116

Mr. James M. Shivley
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 Richland, Washington 99352

Dear Mr. Shivley:

An error was discovered in report BNWL-808, "Radioactive Mass Transport in FFTF Primary Piping." Corrections can be made as follows:

On page 10, second line from the bottom, change "Area" to 67 ft² from 24 ft². This necessitates additional corrections.

All numbers associated with Section xx' in Table V, page 14, should be multiplied by $67/24 = 2.79$.

In Table VI, all radiation levels at point C are to be multiplied by 2.79, so that the pertinent radiation level become those:

Point	Reactor Startup		Ultimate Operation	
	5 ppm oxygen	10 ppm	5 ppm	10 ppm
Scheme 1				
	0.042 R/hr	0.092	1.64	3.65
Scheme 2				
	0.39	0.57	14.8	32.9
Scheme 3				
	0.064	0.142	2.53	5.63

Mr. James M. Shively

-2-

January 15, 1969

Also, on Page A-1, change numbers in text as follows, beginning the third line from the bottom:

$$\text{Area fraction} = \frac{1}{3} \times \frac{67}{259 + 377 + 547} = 0.018$$

Change 0.00065 to 0.0018.

Curies in Section xx' = $0.0018 \times 31.97 = 0.0579$ curies.

Radiation level is $0.0579 \times 0.170 = 0.0098$ R/hr.

It is important to note that these corrections in no way affect the basic validity or conclusions of the report.

Very truly yours,

Original Signed By

P. L. Hofmann

E. R. Astley - Manager
FFTF Project

ERA:mt

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BNWL-808

RADIOACTIVE MASS TRANSPORT IN FFTF
PRIMARY PIPING

By

W. F. Brehm

Development Department
FFTF Project

August 1968

Approved by

D. L. Condotta

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RICHLAND, WASHINGTON

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RADIOACTIVE MASS TRANSPORT IN FFTF PRIMARY PIPING

W. F. Brehm

INTRODUCTION

When a solid is exposed to flowing nonisothermal sodium, a mass-transfer phenomenon may occur. In this situation, material may be removed from the solid in the high-temperature region of the system or loop and eventually be precipitated from the sodium onto a solid surface in the lower-temperature region. The rate of this mass-transfer is affected by many physical parameters such as temperature, temperature change within the system, flow rates and patterns, and presence of impurities in either solid or liquid.

A liquid-metal-cooled nuclear reactor is an excellent example of such a system. Heat is produced in the core region and carried away by the flowing liquid metal to a heat exchanger. The mass-transfer effect will cause removal of material from the fuel cladding in the core and possible deposition of this material in the cooler regions.

In the Liquid Metal Fast Breeder Reactor program (LMFBR), the choice of material for fuel cladding is austenitic stainless steel while for liquid metal coolant the choice is low-oxygen sodium. At the present level of technology these selections give the maximum combination of safety, reliability, and economics for the program. To be specific, it is within the limits of present technology to purify sodium to the required level (approximately 10 ppm of oxygen) for use as coolant and to obtain stainless steel of the necessary properties. Under proper conditions the mass-transfer rate of the stainless steel in the sodium is low enough so that cladding structural integrity can be maintained.

In the FFTF project, our interest in the mass-transfer reaction arises from the fact that the stainless steel removed from the core will be highly radioactive. Active isotopes such as ^{58}Co , ^{54}Mn , ^{60}Co , and ^{51}Cr will be produced in the cladding from interaction of the neutron flux with atoms in the stainless steel. Consequently, even though the material removal rate from the core is so small that cladding integrity will be maintained, the deposition of this radioactive material in colder regions of the heat removal circuit can give rise to large radiation fields there. Our interest is to predict the factors affecting the deposition of radioactive materials, and to develop necessary control measures.

SCOPE

Much of this report will be concerned with an estimate of the magnitude of corrosion and deposition in the FFTF driver core and heat transport system using presently available information, an evaluation of the consequences of the results, and the limitations of the methods used to obtain them. An evaluation is made of the various types of experiments that can be performed to obtain meaningful data that will enable us to predict location and magnitude of radioactive material deposition in FFTF and future LMFBR reactors.

SUMMARY AND CONCLUSIONS

The factors affecting mass transport of stainless steel fuel cladding by flowing sodium have been reviewed. The actual mechanisms of mass transport of the activated stainless steel cladding material under reactor operating conditions are not well defined. Specifically, little experimental evidence is available on the effect of a high heat flux on the mass-transfer process; and also, the distribution of radioactive isotopes (species) throughout a system in which mass transfer is occurring may not coincide with the gross distribution of mass-transferred material.

With these limitations in mind, we calculated the radiation fields near FFTF heat transfer piping using presently available quantitative data on the mass-transfer rate of stainless steel out of the reactor core. Various reactor operating conditions and assumed material deposition schemes were considered. It was found, for example, that under reactor startup conditions with 5 ppm oxygen, 600 °F core inlet temperature and 900 °F core outlet, calculated radiation levels varied from 1.5 to 3 R/hr near heat transport piping and equipment. If ultimate reactor operating conditions (900 °F inlet, 1200 °F outlet, 10 ppm oxygen) are used, radiation levels increase to a level as high as 280 R/hr. Radiation levels of this magnitude can influence component maintenance philosophy.

Recent experimental evidence obtained at PNL shows, however, that the distribution of radioactive material in test loops does not coincide with the gross distribution of mass-transferred material. This adds another unknown to the already large list of unknown factors that affect radioactive mass transport and deposition. Therefore, calculated radiation fields could be in error by several orders of magnitude. It is evident that an experimental program is needed that will provide better quantitative data on the deposition of radioactive

corrosion products for FFTF and LMFBR programs. Such a program is outlined in this document. The program includes continuing tests with thermal convection and pumped sodium loops, high-flux heater tests, and ultimately in-reactor corrosion tests in FFTF. Construction of a sodium-stainless steel model of the FTR is not recommended because the modeling parameters are too poorly defined at this time to proceed with a realistic model.

DISCUSSION

PRESENT STATUS OF LIQUID-METAL MASS-TRANSFER STUDIES

Two excellent general reviews of work in this field are available.^(1,2) However, even today, the corrosion* mechanism of stainless steel in low-oxygen sodium is poorly understood. Extensive work at General Electric Advanced Products Operations has yielded an empirical equation for corrosion of Type 316 SS by flowing sodium, as well as a semi-quantitative study of preferential corrosion of alloy constituents from stainless steel.^(3,4) [The present reference material for FFTF cladding is Type 316 SS. The only real chemical difference between 304 and 316 that affects this study is the molybdenum contained in 316.]

The General Electric Company reports also mentioned deposition studies (deposition in cold legs of material corroded from hot legs.) Even less is known about deposition mechanisms. An interesting recent finding is that increasing the cold leg deposition rate increases the hot leg corrosion rate.⁽³⁾

Two major drawbacks exist in the application of existing mass-transfer data to our specific problem. First, very little information is available on the behavior of irradiated material in flowing sodium. The distribution of radioactive species in a test loop may not correspond to the gross distribution of material that is transported from hot to cold regions by the corrosion process. Second, little experimental work has been reported under conditions that will be present in a fast reactor; that is, the high heat flux in the core that causes a temperature gradient between the cladding surface and the bulk flowing liquid. Under these conditions, corrosion rates may be different from those previously reported and different corrosion mechanisms may operate.⁽⁵⁾

* "Corrosion" and "mass-transfer" are terms used interchangeably in this report.

FFTF DRIVER CORE CORROSION AND DEPOSITION ESTIMATE

An estimate of corrosion of the cladding in the FFTF driver core was made. From this estimate, the out-of-reactor inventory of activated corrosion products and the radiation level at certain places in the primary heat transfer piping was calculated.

The following initial assumptions were made:⁽⁶⁾

- 16,609 fuel pins, 32 in. exposed to flux, 0.250 in. OD
Total area of cladding exposed to flux-417,429 in.²
- Sodium velocity 26 ft/sec.

The corrosion rate is given by the following equation:⁽³⁾

$$R = V^{0.884} C_o^{1.156} \exp \left\{ 12.845 - \frac{23827}{T + 460} + \frac{2.26}{t + 1} \right\} \quad (1)$$

where R = average corrosion rate, mg/dm²-mo to time t

C_o = oxygen concentration, ppm

V = sodium velocity, ft/sec

T = temperature, °F

t = time in months.

The temperature is different for each point along the fuel pin length, hence the "downstream effect" term present in the original Equation (1) does not apply, as it deals with changes in mass-transfer rate along an isothermal section.

Use of the Equation: The following four cases of varying reactor conditions were analyzed:

1. Reactor startup conditions (600 °F bulk sodium core inlet temperature, 900 °F outlet) with 10 ppm oxygen in sodium.
2. Same as case 1 but with 5 ppm oxygen.
3. Reactor at ultimate operating conditions (900 °F inlet, 1200 °F outlet), 10 ppm oxygen.
4. Same as case 3 but with 5 ppm oxygen.

These conditions are to be analyzed at times of one month, four months, one year, and two years of reactor operation.

Equation (1) was used to calculate corrosion rates for increments along a fuel pin length (which are at different temperatures). These rates were used to obtain an average corrosion rate for an entire fuel pin length. An average temperature was then back calculated from this corrosion rate. A 12 °F bulk-to-cladding temperature gradient was assumed for the reactor startup case, a 30 °F gradient was assumed for the ultimate operation case.⁽⁷⁾ The combination of these two steps led to an "average" fuel cladding surface temperature of 800 °F in the reactor startup case, and 1100 °F for the ultimate operation case. With these considerations, values were calculated for cladding corrosion using the Equation (1). They are given in Table I.

TABLE I. *Corrosion Rates of Fuel Cladding, mg/dm²-mo. for Various Operating Times*

<u>Reactor Startup Conditions</u>		
<u>(600 °F Inlet, 900 °F Outlet)</u>		
<u>Time</u>	<u>10 ppm oxygen</u>	<u>5 ppm oxygen</u>
1 mo	1.834	0.834
4 mo	0.931	0.418
1 yr	0.705	0.316
2 yr	0.649	0.291
<u>Ultimate Reactor Conditions</u>		
<u>(900 °F Inlet, 1200 °F Outlet)</u>		
<u>Time</u>	<u>10 ppm oxygen</u>	<u>5 ppm oxygen</u>
1 mo	69.63	31.24
4 mo	35.35	15.86
1 yr	26.76	12.01
2 yr	24.62	11.04

The corrosion rates were converted to an out-of-reactor curie inventory in the following way. The rates in $\text{mg/dm}^2\text{-mo.}$ were converted to cm^3 of cladding removed per day. These values are given in Table II. The concentration of active species in Ci/cm^3 of cladding material was taken from available information.⁽⁸⁾ The four isotopes considered were these:

^{60}Co , 2 Ci/cm^3 , 1913 days half-life
 ^{51}Cr , 3 Ci/cm^3 , 27.8 days half-life
 ^{58}Co , 31 Ci/cm^3 , 71 days half-life
 ^{54}Mn , 10 Ci/cm^3 , 313 days half-life

^{59}Fe is not considered because it is expected to be produced in very small quantities in FFTF cladding. ^{56}Mn and ^{99}Mo , which will be formed by irradiation of the cladding, are not considered because of their short half-lives, (2.6hr and 67 hr, respectively).

TABLE II. Average Amount of Material Removed from Core Area, cm^3/day for Various Operating Times

<u>Reactor Startup Conditions</u>		
<u>Time</u>	<u>10 ppm oxygen</u>	<u>5 ppm oxygen</u>
1 mo	0.2112	0.0948
4 mo	0.1072	0.0481
1 yr	0.0812	0.036
2 yr	0.0747	0.0334
<u>Ultimate Operating Conditions</u>		
<u>Time</u>	<u>10 ppm oxygen</u>	<u>5 ppm oxygen</u>
1 mo	8.02	3.60
4 mo	4.07	1.83
1 yr	3.08	1.383
2 yr	2.84	1.27

The preferential leaching of chromium and manganese⁽⁴⁾ is not considered, since no quantitative information is available. Estimates of the solubility of cobalt in flowing sodium are also scarce.

The information in Table II can be then converted into out-of-reactor curie levels with the following equation:

$$C = \frac{K}{\lambda} \left\{ 1 - (1 + \lambda t) \exp(-\lambda t) \right\} \quad (2)$$

where C = amount of active species present at time t, Ci

$$K = \frac{\text{curies}}{\text{cm}^3} \text{ at saturation } \times \text{cm}^3 \frac{\text{transported}}{\text{day}}$$

$$\lambda = \text{decay constant, (days)}^{-1}$$

$$= \ln 2 / T_{1/2} \text{ where } T_{1/2} \text{ is in days}$$

$$t = \text{time, days}$$

This equation has been developed by Haag⁽⁹⁾ and can be obtained from equations derived by Bunch and Zweibel.⁽¹⁰⁾ It is evident that an eventual out-of-reactor saturation occurs, but only after long times and at high curie levels with long-lived species. Curie levels for the four isotopes considered are given in Table III; a typical plot of out-of-reactor buildup of isotopes is shown in Figure I.

Radioactive Material Deposition in Heat Transfer Piping

A schematic model of a typical FFTF primary heat transfer loop was assumed,^(11,12) as shown in Figure 2.

The following information and assumptions were used:

1. No deposition occurs in the reactor vessel or upstream of the heat exchanger. This is reasonable if there is no temperature drop upstream of the inlet of the heat exchanger.
2. Pipe sizes and areas are as shown in Table IV and Figure 2.

TABLE III. *Out-of-Reactor Activity, in Curies,
As Function of Time and Isotope*

<u>Reactor Startup Conditions, 5 ppm Oxygen</u>				
<u>Time</u>	<u>^{60}Co</u>	<u>^{51}Cr</u>	<u>^{58}Co</u>	<u>^{54}Mn</u>
1 mo	0.0307	1.97	10.64	0.907
4 mo	0.2438	4.627	49.95	6.453
1 yr	1.609	4.374	100.6	31.97
2 yr	5.42	4.018	105.36	72.47
<u>Startup Conditions, 10 ppm Oxygen</u>				
1 mo	0.684	4.39	23.71	2.02
4 mo	0.543	10.31	111	14.58
1 yr	3.58	9.75	224	71.24
2 yr	12.08	8.95	235	161.49
<u>Ultimate Operation, 5 ppm Oxygen</u>				
1 mo	1.17	74.8	404	34.4
4 mo	9.27	176	1900	246
1 yr	61.15	166	3823	1214
2 yr	206	153	4006	2755
<u>Ultimate Operation, 10 ppm Oxygen</u>				
1 mo	2.607	166.6	900	76.7
4 mo	207	392	4233	548
1 yr	136	370	8519	2705
2 yr	459	341	8926	6139

TABLE IV. *Heat Transport Loop Sizes^(a)*

<u>Component</u>	<u>Length</u>	<u>Pipe Size</u>	<u>Area</u>
IHX			3810 ft ²
IHX to pump	43 ft	24 in. OD, 0.5 in. wall	259
Pump			377
Pump to inlet	123	18 in. OD, 0.5 in. wall	547
Section XX'	15	18 in. OD, 0.5 in. wall	24
Total, one loop			4993 ft ²

(a) (See Figure 2)

FIGURE 1. Buildup of Out-of-Reactor Activity,
Startup Conditions at 5 ppm Oxygen

Points A, B, C are two feet away from wall at midpoint of component, XX' is 15 ft.

Radiation at B from pump only

Pipe sizes, etc., in Table IV

IHX, right circular cylinder 20 ft high, 3 ft OD, 0.25-in. case, 25% of enclosed volume is stainless steel

Pump, right circular cylinder 40 ft high, 3 ft OD, 0.25-in. case

Deposition on inner case wall

FIGURE 2. Schematic Drawing of Heat Transport System
(Drawing SK-3-12750 in BNWL-554)

There are four similar heat transfer loops in the FFTF. Plans are to use three of them at a time. Therefore, in estimating deposition levels in each loop the curie levels of Table III are to be divided by three, then scaled to the pertinent area fraction according to these assumed deposition schemes:

1. 90% of these activation products deposit in heat exchanger, 10% deposit uniformly over area downstream from IHX (including pump).
2. 10% deposit in IHX, 90% over downstream area. This will occur if the sodium becomes supersaturated with corrosion products prior to deposition.
3. 10% in heat exchanger, 80% in pump, 10% over the rest of the piping system. This situation may occur if flow patterns in the pump favor deposition.

It must be emphasized that these deposition schemes were derived from speculation and are not based upon any experimental evidence.

Table V shows curies of deposited active material according to these schemes after one year of operation.

It is now necessary to translate these curie levels into radiation levels at the chosen locations (A, B, C on Figure 2). C. A. Mansius of the FFTF Shielding Analysis Unit calculated radiation levels per curie of isotope deposition at points A, B, and C, using computer code ISOSHLTD. This code calculates radiation levels for species uniformly mixed in the sodium. It was assumed that in the IHX the values for deposition on tube walls would not be greatly different from those calculated from ISOSHLTD. However, in the pump and piping the deposition was assumed to be on the walls, therefore, the values are approximations.

TABLE V. *Curies Deposited in Certain Piping Sections at 1 Year Time*

Scheme 1 (90% in IHX, 10% Elsewhere)

<u>Operating Conditions</u>		<u>60Co</u>	<u>51Cr</u>	<u>58Co</u>	<u>54Mn</u>
Startup	IHX	0.483	1.31	30.18	9.59
5 ppm oxygen	P	0.0164	0.0446	1.026	0.326
	XX'	0.001	0.0028	0.065	0.0207
Startup	IHX	1.07	2.91	67.2	21.3
10 ppm oxygen	P	0.037	0.099	2.28	0.726
	XX'	0.002	0.006	0.144	0.046
Ultimate Operation	IHX	18.34	49.8	1146	364
5 ppm oxygen	P	0.623	1.693	39	12.4
	XX'	0.040	0.1078	2.48	0.789
Ultimate Operation	IHX	40.8	111	2553	811
10 ppm Oxygen	P	1.39	3.77	87	27.6
	XX'	0.089	0.240	5.52	1.758

Scheme 2 (10% in IHX, 90% Elsewhere)

Startup	IHX	0.0536	0.1458	3.35	1.06
5 ppm oxygen	P	0.146	0.398	9.17	2.91
	XX'	0.009	0.025	0.5885	0.187
Startup	IHX	0.119	0.324	7.46	2.36
10 ppm oxygen	P	0.325	0.886	20.3	6.48
	XX'	0.020	0.055	1.31	0.416
Ultimate Operation	IHX	2.03	5.53	127	40.4
5 ppm Oxygen	P	5.57	15.13	348	111
	XX'	0.3577	0.971	22.36	7.1
Ultimate Operation	IHX	4.52	12.3	283	90
10 ppm Oxygen	P	12.4	33.7	775	247
	XX'	0.797	2.16	49.8	15.8

Scheme 3 (10% in IHX, 80% in Pump, 10% Elsewhere)

Startup	IHX	0.0536	0.1458	3.35	1.06
5 ppm Oxygen	P	0.429	1.16	26.8	8.52
	XX'	0.0015	0.0043	0.100	0.032
Startup	IHX	0.119	0.324	7.46	2.46
10 ppm Oxygen	P	0.995	2.58	59.7	18.98
	XX'	0.003	0.009	0.223	0.071
Ultimate Operation	IHX	2.03	5.53	127	40.4
5 ppm Oxygen	P	16.3	44.2	1019	323
	XX'	0.062	0.166	3.823	1.214
Ultimate Operation	IHX	4.52	12.3	283	90
10 ppm Oxygen	P	36.3	98.4	2270	719
	XX'	0.138	0.369	8.51	2.71

The calculated values are:

<u>Isotope</u>	<u>(R/hr)/Curie Deposited</u>		
	<u>Point A</u>	<u>Point B ($\pm 40\%$)⁽¹³⁾</u>	<u>Point C ($\pm 25\%$)⁽¹³⁾</u>
⁶⁰ Co	0.026	0.255	0.530
⁵¹ Cr	0.00064	0.003	0.005
⁵⁸ Co	0.027	0.091	0.175
⁵⁴ Mn	0.026	0.0905	0.170

It now remains to multiply the values of Table V by the appropriate conversion value above, and total the radiation levels. The results are shown in Table VI.

A sample calculation is found in the Appendix.

In Table VI it is demonstrated that a wide spectrum of radiation levels is produced depending on operating and deposition conditions. The high radiation levels near the IHX are of no concern since the IHX is to be designed for remote maintenance.⁽¹¹⁾ However, the radiation fields produced near the pump and piping even during initial reactor operation may have an impact on the maintenance philosophy of these components. Much of the deposited activity is from 71 day ⁵⁸Co and 313 day ⁵⁴Mn, so that the high radiation level will be present for a long time after a loop is shut down.

Other Situations

1. Deposition in the Reactor Vessel. If the deposition does not take place in the heat transport piping, a large amount of activated corrosion products will flow into the bottom of the reactor vessel, and perhaps be deposited there. This would represent an ideal case, since then any problem of radiation levels from deposition near heat transfer loops would be reduced.

TABLE VI. *Radiation Levels at Selected Locations
After 1 yr Reactor Operation*

	<u>Reactor Startup</u>		<u>Ultimate Operation</u>	
	<u>5 ppm oxygen</u>	<u>10 ppm oxygen</u>	<u>5 ppm oxygen</u>	<u>10 ppm oxygen</u>
	<u>Scheme 1 (90% in IHX, 10% Elsewhere)</u>			
Point A (IHX)	1.10 R/hr	2.45	42.1	93.8
B (pump)	0.127	0.283	4.84	10.7
C (XX')	0.015	0.033	0.589	1.31
	<u>Scheme 2 (10% in IHX, 90% Elsewhere)</u>			
Point A	0.122	0.271	4.67	10.4
B	1.136	2.53	43.2	96.1
C	0.140	0.312	5.31	11.8
	<u>Scheme 3 (10% in IHX, 80% in Pump, 10% Elsewhere)</u>			
Point A	0.122	0.271	4.67	10.4
B	3.32	7.39	126	281
C	0.023	0.051	0.909	2.02

2. Trapping. General Electric Company reported increased deposition near flow perturbations in their test loops.⁽³⁾ A number of these (e.g. bends, tees, elbows, check valves) will be present in heat transport piping.^(11,12) In addition, other work has shown that increased deposition of magnetic species occurred near flowmeters.⁽¹⁴⁾ With the large ⁵⁸Co inventory present, this must be considered. As mentioned before, little quantitative information is available on this subject but studies are underway at Battelle-Northwest, General Electric-Advanced Products Operation, and Mine Safety Appliance Research Corporation.
3. Preferential Leaching, Effect of Radiation. The data presented were obtained ignoring the known preferential leaching of Cr and Mn that occurs with stainless steel in low-oxygen sodium. The effect of neutron flux on sodium corrosion of stainless steel has also been ignored. Again, lack of data necessitates this approach.

4. Fission Product Release. A release of soluble fission products to the sodium would present a situation similar to the corrosion case. This raises the question, are the elements being deposited, and if so, where? The physical and chemical principles involved should be similar to those just discussed. Some aspects of a fission-product release have been treated.⁽¹⁵⁾
5. Activity Transport Not Coincident with Mass Transport. Preliminary data obtained in the Radioisotope Transport Loop at PNL (where an irradiated stainless steel specimen was placed in the flowing sodium at the highest-temperature location) show that the principal radioactive species released into the sodium (^{51}Cr in this case) had not migrated to the cold leg as was previously predicted, but was distributed near the specimen location and immediately downstream. Apparently an isotopic exchange mechanism was occurring between radioactive atoms released from the specimen to the sodium and inert atoms on the pipe walls. This phenomenon is superimposed on the gross mass-transfer of material taking place.

These results were confirmed in a thermal convection loop test using an irradiated stainless steel specimen. The specimen showed a weight loss about equal to that calculated from the General Electric Company equation for the loop operating parameters, but the largest amount of activity was found deposited within a 6 in. region immediately downstream of the specimen. The relative "spreading-out" of the activity was slightly larger in the thermal convection loop, but the general pattern of decreasing activity count with increasing downstream distance was similar. This means that the assumption previously listed, that no radioisotope deposition occurs upstream of the heat exchanger inlet, may not be valid.

These results may have major significance upon the actual amount of radioactive species released to the heat-transfer piping in FFTF. Present FFTF core design plans are for a 3 ft gas plenum above the fuel pellets inside each fuel pin. This means that a large area of stainless steel will be located immediately downstream from the core outlet. If the isotopic exchange process occurs there, the majority of the activated corrosion products will not leave the fuel pin cladding just above the core. The ones that are not trapped there may be trapped at the reactor outlet nozzle, or plate out on the reactor vessel. In any case, the high radiation levels anticipated to be present near heat exchanger piping will be reduced.

This following point must also be considered: If an exchange mechanism is actually taking place, activity deposited immediately downstream from the core outlet may eventually be transported out of the reactor vessel by continuing exchange processes. The time-dependence of this phenomenon is not known, but it should be investigated.

CONCLUSIONS

From the discussion in the previous section, it is apparent that a large inventory of radioactive species will be released to the coolant during FFTF operation, and that we do not have, at this time, the knowledge to predict its deposition behavior in the FFTF heat transfer piping. With such a large inventory, it is essential to be able to estimate location and magnitude of this deposition, rather than rely upon the approximations used in making the calculations, especially when the first experimental evidence contradicts some of the assumptions.

As mentioned before, experimental programs for investigation of interactions between sodium and stainless steel are

underway at Battelle-Northwest, General Electric-Advanced Products Operation, Atomics International, and Brookhaven. The program at Battelle is most closely related to the FFTF operation. We believe that the following information should be available by the time the FFTF is ready for operation: (1) a reasonable, but not necessarily exact, estimate of the corrosion rate of stainless steel exposed to a high heat-flux and flowing high-velocity sodium; (2) knowledge of the flow patterns of the coolant through the FFTF core and primary heat transfer piping; (3) information about magnitude and location of deposition of active species in the core and piping; and (4) information about the "mobility" of deposited radioactive atoms, especially those deposited in locations other than that predicted from gross mass transfer data.

Some experimental techniques to obtain this information include the following:

1. Thermal convection loops. These can be of variable geometry to study flow-pattern effects, are easily set up, heated, and scanned for radioactive deposits, and are also relatively inexpensive. However, the velocity obtained in them is so low that prototypical corrosion rates cannot be obtained. They are ideal to use for the study of the "mobility" problem.
2. Pumped Loops (such as the Radioisotope Transport Loop at Battelle-Northwest). These can become complicated and expensive; but permit close control over temperature, flow rate, and impurity concentration. The radioisotope loop is very adaptable to changes in test section geometry.
3. Hydraulic Core Mockup of FFTF Core. This facility will permit study of flow patterns in the reactor vessel. The problem of adequate evaluation of model criteria seems to be under control.⁽¹⁶⁾

4. Heated Sodium-Stainless Steel Core Mockup. This facility would permit study of flow patterns and activity deposition under model conditions. However, not enough is known about the behavior of active species in sodium to warrant expense of building this facility. More information can be gained by correlating information from thermal convection loops, pumped loops, and the hydraulic core mockup, so tests with a heated sodium-stainless steel core mockup are not recommended.
5. High-Heat-Flux Tests. These tests will provide meaningful data on the source term in the mass-transfer process, and should be continued. At the present time, Brookhaven and AI are the laboratories most closely associated with this effort.

It is therefore recommended that the Battelle-Northwest program consist of maintaining the present effort under Task RS-9 (to become RM-12) with thermal convection and pumped loops. Close liaison should be maintained with other sites, especially those involved with study of high-heat-flux conditions. An experiment that should be initiated soon is the insertion of a high-flux heater (clad with an irradiated stainless steel sheath) into a pumped, high-velocity sodium loop.

In-Reactor Corrosion Studies

The best place to study in-reactor corrosion and deposition in the presence of neutron flux, high heat flux, and complex flow patterns is in the reactor itself. A test facility such as FFTF, with the planned close control over flow, temperature, flux, and impurity level would be an ideal reactor in which to do this. Emphasis should be given to surveillance of radioactivity deposited in certain sections of FFTF core and closed loop heat transport piping as planned in Task SG-7. This study is of obvious value to the LMFBR program.

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APPENDIX

SAMPLE CALCULATION

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SAMPLE CALCULATION

Consider ^{54}Mn buildup for 12 months (365 days) in case of 5 ppm oxygen, reactor startup (600 °F inlet, 900 °F outlet). For this case, $T_{\text{avg}} = 800$ °F. Calculate corrosion rate:

$$R(\text{mg/dm}^2\text{-mo.}) = (26)^{0.884} \times (5)^{1.156} \\ \times \exp \left\{ 12387 - \frac{23827}{800 + 460} + \frac{2.26}{12 + 1} \right\} \\ = 0.316 \text{ mg/dm}^2\text{-mo.}$$

This must be converted to cm^3/day of cladding removed by multiplying by

$$0.001 \frac{\text{g}}{\text{mg}} \times 0.01 \frac{\text{dm}^2}{\text{cm}^2} \times 6.54 \frac{\text{cm}^2}{\text{in.}^2} \times 417429 \text{ in.}^2 \times \frac{1 \text{ cm}^3}{7.9 \text{ g}} \\ \times \frac{1 \text{ mo.}}{30 \text{ day}} = 0.1152 \frac{\text{cm}^3/\text{day}}{\text{mg/dm}^2\text{-mo.}}$$

$$\text{So } \frac{\text{cm}^3}{\text{day}} \text{ removed} = 0.1152 \times 0.316 = 0.0364 \frac{\text{cm}^3}{\text{day}}$$

$$\text{Out-of-reactor } ^{54}\text{Mn curie level} = 0.0364 \frac{\text{cm}^3}{\text{day}} \times 31 \frac{\text{Ci}}{\text{cm}^3}$$

$$\times \frac{313 \text{ days}}{0.693} \times \left[1 - \left(1 + \frac{365 \times 0.693}{313} \right) \exp - \left(\frac{365 \times 0.693}{313} \right) \right]$$

$$= 31.97 \text{ Ci.}$$

In section XX' according to Scheme I

$$\text{Area fraction} = \frac{1}{3} \times \frac{24}{(259 + 377 + 547)} = 0.0065$$

With 10% of corrosion species depositing in piping downstream from IHX, pertinent fraction is 0.00065.

Curies in section XX' - $0.00065 \times 31.97 = 0.0207$ curies. ^{54}Mn deposited in section XX' yields 0.170 R/hr at point C per curie. Therefore, radiation level is $0.0207 \times 0.170 = 0.0035$ R/hr. Table VI was obtained by summing values for the four isotopes for each deposition scheme.

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