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SODIUM TECHNOLOGY
TECHNICAL PROGRESS REPORT
JULY, AUGUST, SEPTEMBER
1975

MASTER

HANFORD ENGINEERING DEVELOPMENT LABORATORY
Operated by Westinghouse Hanford Company
A Subsidiary of Westinghouse Electric Corporation

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J. M. Atwood

February, 1976

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Hanford Engineering Development Laboratory

Operated by the
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ABSTRACT

This report was prepared at the Hanford Engineering Development Laboratory operated by Westinghouse Hanford Company, a subsidiary of Westinghouse Electric Corporation, for the United States Energy Research and Development Administration, Division of Reactor Research and Development, under Contract No. E (45-1) 2170. It describes technical progress made during the reporting period by the Sodium Technology Department within Westinghouse Hanford Company.

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INTRODUCTION

This is the third in a series of Sodium Technology Quarterly reports. The initial report, HEDL-TME 75-3, covered the first three months of 1975, and the second report, HEDL-TME 75-64, covered the second quarter of 1975. Previously, Sodium Technology progress was incorporated into the HEDL Quarterly Technical Reports.

Progress is reported by the four organizational components within Sodium Technology:

- Radioactivity Control Technology
- Sodium Systems Development
- Sodium Systems Engineering
- Sodium Systems Analysis

Major topics covered include:

- Corrosion product release, transport, and deposition
- Fuel and Fission Product Transport and Deposition
- Tritium control in LMFBFR's
- Decontamination of LMFBFR components
- Design and procurement of the FFTF Sodium Characterization System and initial work on SCS for CRBR
- Design and performance analysis of FFTF cold traps
- Sodium removal process development
- Liason with off-site sodium technology programs
- Investigation of flow impedance phenomena
- Chemical analysis and instrumentation
- Development of a protective suit for sodium emergency personnel
- Fuel failure monitoring systems for FFTF and CRBR

RADIOACTIVITY CONTROL TECHNOLOGY

R. J. Cash

The Radioactivity Control Technology (RCT) Program at HEDL is developing methods and base technology to predict, measure, control, and remove radioactivity from the heat transport system of an LMFBR. The radionuclides of interest are the corrosion products (chiefly ^{60}Co , ^{58}Co , and ^{54}Mn), the various fission products released when operation with failed fuel is permitted, and tritium which is also generated in the core and diffused into the coolant circuits. Because of the complexity and magnitude of radionuclide transport and distribution, several methods are under investigation.

At LMFBR operating temperatures, radioactivity from corrosion product transport and deposition alone will be sufficient to severely limit access time for maintenance of system components unless control measures are instituted. Fission product releases from failed fuel operation will add to this burden significantly unless technology can be developed to limit their release and subsequent deposition in critical maintenance areas.

This report discusses work currently underway and planned as part of the RCT program.

Improved Model for Corrosion Product Transport - W. L. Kuhn

Objective

To identify and characterize potential maintenance problems of the FFTF Heat Transport System (HTS), the rate of release of activated corrosion products and fission products from the core and their subsequent accumulation in the HTS piping and components must be estimated. This task provides an improved model for estimating corrosion product and fission product activity levels in the FFTF Heat Transport System.

Summary of Accomplishments

- Progress this quarter has been limited to an analysis of activated corrosion products. Fission product burdens will be analyzed at a later date.
- The computation of the release of activated corrosion products from the FFTF core has been refined. The new results are combined with previous estimates of the distribution of radionuclide inventories over the FFTF primary coolant loops and shielding computations to provide a revised estimate of the radiation dose rates at some key maintenance locations in the HTS cell. The results are presented in Tables 2, 3, and 4, and Figure 2. The dose rates for a 600⁰F core inlet are found to be twenty times smaller than for an 800⁰F core inlet temperature. This result depends on the activation energies of the important radionuclide release mechanisms, which should be checked experimentally to increase confidence in the lower temperature result.

Discussion

Introduction

The computation of radiation dose rates within an HTS cell is divided into three operations: (1) the computation of the inventories of radionuclides released from the core, (2) the computation of the distribution of the inventories over the sodium wetted surfaces of the primary coolant loop, and (3) the computation of the cumulative dose rate due to the distributed inventories at given dose points. The refinement of any of these operations thus changes the estimates of dose rates.

A major activity of the RCT program over the past three years has been the improved definition of radionuclide release rates from 316 SS fuel cladding during corrosion in high velocity sodium at several levels of oxygen concentration and a single temperature, 1120⁰F, the average cladding temperature for ultimate FFTF operation. Supplemental information regarding the temperature variation of important release phenomena have become available in the literature. Accordingly, this task has been directed principally toward the refinement of the radionuclide release computations.

Release Computations

The computational technique is described in Reference 1, and the results are presented in Reference 2. A brief discussion of these is included here for continuity.

The release of the radionuclides is controlled mostly by two phenomena: surface corrosion and solid-state diffusion. In the absence of diffusion, the concentration of a radionuclide is uniform throughout the cladding, so that the release rate is the product of the surface corrosion rate and the uniform concentration. However, if the release of a nuclide tends to occur faster or slower than the average release of the stainless steel components, then the accumulation or depletion of the radionuclide near the surface is accommodated by solid state diffusion. The species diffuses out of or back into the base metal in order to maintain a mass balance at the surface.

These phenomena can be represented by a partial differential equation describing the radionuclide mass transport in the cladding, with appropriate boundary conditions. The equation is solved to obtain the concentration profile, from which the radionuclide release rates and ultimately the inventories are obtained.

Dose Rate Computations

Having refined the release calculations, the inventory distribution and shielding computations of Reference 3 are used to obtain a new estimate of dose rates at key maintenance locations in the HTS cell (Figure 1). Three locations were chosen: (1) 2 feet above the cold-leg check valve, (2) a location at the cold leg near temperature and pressure transducers and the flowmeter, and (3) a location at the hot leg near an expansion bellows.

The radionuclide inventories were computed for three reactor operating conditions, shown in Table 1.

HEDL 7210-23.11

FIGURE 1. FFTF Heat Transport Cell Piping and Component Layout Showing Locations Where Dose Rates were Calculated.

FIGURE 2. Dose Rates in HTS Cells for Various Operating Conditions of FFTF.

TABLE 1

REACTOR OPERATING CONDITIONS

<u>Case</u>	<u>Core Inlet, °F</u>	<u>Oxygen Level, ppm</u>
A	800	0.5
B	800	2.5
C	600	0.5

The dose rates at locations (1), (2), and (3) are presented in Tables 2, 3, and 4, respectively. The dose rates for locations (2) and (3) are presented in Figure 2. Note that the dose rates for Case C at the lower operating temperature are about 20 times smaller than for the "design" condition, Case A. This result depends on a knowledge of the activation energies of the corrosion and diffusion processes, since the computations are based largely on data taken at temperatures more representative of Case A than Case C. Therefore, an experimental determination of the corrosion and diffusion rates at temperatures near 900°F would be more desirable.

Expected Accomplishments Next Quarter

- Additional computer calculations will be completed to predict radiation dose levels at other locations in the HTS cell. This work will culminate in a report updating Reference 3.

Radioactive Corrosion Product Studies - W. F. Brehm, J. C. McGuire,
L. E. Chulos, R. R. Rake, and J. L. Bevan

Objective

The objective of this work is to characterize the release and transport of corrosion products in flowing sodium as a function of system operational parameters. The results of this study will help to specify LMFBR temperature and coolant purity limitations if required, and also to predict nuclide deposition locations of nuclides in primary heat transport systems. The data from this study are used in the improved analytical model for corrosion product transport discussed previously.

TABLE 2.

DOSE RATES AT LOCATION (1), R/HR

<u>Years</u>	<u>Case A</u>	<u>Case B</u>	<u>Case C</u>
.27	.79	.97	.04
.64	1.80	2.27	.09
1.00	2.68	3.40	.14
2.08	4.17	4.34	.21
5.34	5.21	6.74	.26
10.8	5.34	6.98	.27

TABLE 3.

DOSE RATES AT LOCATION (2), R/HR

<u>Years</u>	<u>Case A</u>	<u>Case B</u>	<u>Case C</u>
.27	1.03	1.32	.05
.64	2.31	3.02	.12
1.00	3.40	4.47	.17
2.08	5.27	6.95	.26
5.34	6.60	8.82	.33
10.8	6.83	9.27	.34

TABLE 4.

DOSE RATES AT LOCATION (3), R/HR

<u>Years</u>	<u>Case A</u>	<u>Case B</u>	<u>Case C</u>
.27	.31	.45	.02
.64	.67	.98	.03
1.00	.97	1.40	.05
2.08	1.48	2.11	.07
5.34	1.88	2.73	.09
10.8	1.99	2.98	.10

Summary of Accomplishments

- Data from the interim shutdown of STCL-1 after 3000 hours at 1120°F with a hot trap show weight losses and radionuclide releases larger than were observed at 0.5 ppm oxygen, indicating that purification by hot trapping is an ineffective method for controlling ^{54}Mn release.
- The 316 SS specimens in STCL-2 and 3 sustained weight losses in good agreement with the expected results for 3000 hours at 0.5 ppm oxygen.
- STCL-1, 2, and 3 were restarted after interim shutdowns as shown in the test matrix (Table 9).

Discussion

Results of Corrosion Testing

STCL-1 was shutdown for an interim 3000-hour examination during this quarter. During the test, the oxygen level was 0.7 to 2 parts per billion (ppb) as indicated by oxygen meter readings near 1.905 volts. (Calibration procedures have been described previously in Reference 4). The corrosion specimens were removed, weighed, and the unirradiated ones were gamma-scanned. One irradiated 316 SS specimen was used for incremental analysis. The remaining specimens, the replacement irradiated 316 SS specimen and a new unirradiated specimen to fit between the tapered nosepiece and the first irradiated specimen, were remounted and re-inserted into the STCL-1 loop. Test conditions were resumed (1120°F, 22 ft/sec velocity) for the final 3000 hours of the low oxygen corrosion test. The oxygen level has remained at 0.7 to 2 ppb, based upon a fairly steady oxygen meter reading of 1.900 to 1.910 volts.

Weight-loss rates were much higher than expected, as shown in Tables 5 and 6. The corrosion rates were much higher than obtained in the previous 0.5 ppm oxygen test^(1, 4). It can be seen in Tables 5 and 6 that significant differences in corrosion rates were obtained between the various groups of specimens. Part of the decreasing corrosion rate along the length of the specimen holder can be attributed to the "downstream effect"⁽⁵⁾. However, this effect cannot account for the step changes between the "E" series of

TABLE 5

STCL-1 SPECIMEN WEIGHT LOSS DATA
3000 HOURS, 1120°F, 22 FT/SEC, <0.01 PPM OXYGEN

Specimen	$\Delta W \text{ } \mu\text{g/mm}^2$	Equivalent Corrosion Rate mils/yr	microns/year
(R) A49-CW 316SS	9.1	0.13	3.3
E2	8.3	0.12	3.0
(R) E3	7.7	0.11	2.8
E4	6.84	0.10	2.5
E5	7.87	0.11	2.9
16B ⁺	5.13	0.074	1.9
17B	5.4	0.078	2.0
18B	6.0	0.087	2.2
19B	6.17	0.089	2.3
B94 [#] SA	3.48	0.050	1.3
B62 CW	6.46	0.093	2.4
B91 SA	3.80	0.055	1.4
B51 CW	5.36	0.077	2.0

↓
flow

⁺ 16B-19B are 316SS but with 0, 1, 2, 4% Mn respectively

[#] SA - solution annealed 1950°F 1 hr, quenched

[#] CW - cold worked

R - replaced for second 3000 hours

TABLE 6.

AVERAGED STCL-1 SPECIMEN WEIGHT LOSS DATA
 3000 HOURS, 1120°F, 22 FT/SEC, <0.01 PPM OXYGEN

	$\Delta W \text{ } \mu\text{g/mm}^2 *$	Equivalent Corrosion Rate <u>mils/yr</u>	<u>microns/year</u>
All specimens	6.28	0.01	2.3
Irradiated 316SS (E2, 3, 4, 5)	7.67	0.11	2.8
16B-19B	5.68	0.082	2.1
Cold worked unirradiated (except A49)	5.91	0.085	2.2
S.A.	3.64	0.052	1.3
Average 3000 hour data at 0.5 ppm oxygen (irradiated 316SS) (see Reference 4)	4.71	0.068	1.73

*The relationship in Reference 4 is

$\Delta W (\mu\text{g/mm}^2) = 1.15t + 1.26$ where t is measured in 1000-hour increments for 1120°F, 22 ft/sec, 0.5 ppm oxygen.

irradiated 316 SS and the 16B-19B series nor the difference in corrosion rates observed between the solution-annealed specimens and the cold-worked specimens.

Incremental analysis was done on specimen E3; the analysis showed that large concentration gradients in Ni, Cr, Mn, and ^{54}Mn extended more than 0.2 mils into the base material. ^{60}Co and Mn were retained in the material and the surface was enriched in Fe. Table 7 shows a comparison of corrosion rates for each element and nuclide for 3000-hour specimens at <0.01 ppm oxygen and 0.5 ppm oxygen. The comparison shows that the iron corrosion rate was about the same in the two specimens, while substantial increases in Cr, Ni, Mn, ^{54}Mn , and ^{60}Co releases are observed in the specimen tested at <0.01 ppm oxygen. Figure 3 shows the depleted zone for ^{54}Mn to be much larger in the <0.01 ppm specimen. The deeper "depleted zone" accounts for the much larger ^{54}Mn release observed in the <0.01 ppm test. A much more detailed comparison (of the element and nuclide release rates) will be made at the conclusion of the 6000-hour run at <0.01 ppm oxygen when all six irradiated specimens will be analyzed.

It has been hypothesized in the past⁽⁶⁾ that the corrosion rate of stainless steel in oxygen-free sodium would be higher than at 1 to 10 ppm oxygen because of the lack of an iron-sodium oxide protective layer or layer of adsorbed oxygen on the steel surface to act as a buffer. Under "low-oxygen" conditions it has been hypothesized that corrosion would occur by simple solubility processes and would be much greater, based upon reported differential solubilities of iron at loop hot leg and cold leg temperatures^(6, 7, 8).

It is not possible to evaluate these hypotheses using the limited amount of data in Table 7; however, it is worth noting that a large part of the corrosion weight loss for both specimen E3 (<0.01 ppm) and 31C (0.5 ppm) was Ni and Cr. A substantial portion of the Ni and Cr losses occurs by diffusion out of the metal matrix. It is hardly conceivable to expect Fe, Cr, Ni, and Mn to all respond in the same manner to changes in oxygen in sodium. The change in Ni, Cr, and ^{54}Mn release rates must be explained by a change in the effective diffusion coefficient within the material and the

TABLE 7.

WEIGHT LOSS ($\mu\text{g}/\text{mm}^2$) BY INCREMENTAL ANALYSES OF IRRADIATED
316 SS EXPOSED FOR 3000 HOURS AT 1120°F IN STCL-1

	<u>Specimen E3</u> <u>(hot trap, <0.01 ppm oxygen)</u>	<u>Specimen 31C</u> <u>(0.5 ppm oxygen-Run 6)</u>
Total	7.7	5.5 ⁺
Fe	1.12	1.27
Ni	2.62	1.69
Cr	3.26	1.69
Mn	0.44	0.3
Mo.	0.26	0.57
⁵⁴ Mn release	1.02% (of total specimen inventory)	0.57%
⁶⁰ Co release	0.18%	0.12%

⁺ A value of $4.72 \mu\text{g}/\text{mm}^2$ is predicted from the correlation given in Reference 4 for 0.5 ppm oxygen, 3000 hours at 1120°F. It is important to note that the Reference 5 weight-loss correlation, as well as the data used in the E3-31C comparison above, were obtained on specimens with nearly identical processing history and in the same test loop (STCL-1).

FIGURE 3. Relative ^{54}Mn Depleted Zones After 3000-Hours Exposure For Three Oxygen Levels.

HEDL 7602-57.20

tendency to form ferrite on the surface. The ferrite will have a diffusion coefficient for substitutional atoms of at least an order of magnitude higher than austenite at the same temperature. From Figure 3 it appears that the <0.01 ppm oxygen specimen (E-3) formed ferrite on the surface to a greater extent than the other specimens--based upon the flat diffusion profile for the first two increments.

Gamma-Scanning STCL-1

During the interim 3000-hour shutdown, the insulation was removed from the STCL-1 pipework for gamma-scanning. This scan gave distribution patterns in the piping after 18,000 hours operation, except in the new economizer, where 3000-hour data were obtained. The results were in general accord with those observed throughout this program. Therefore, the observations listed below are to be taken as the summary for 18,000 hours' observation and seven separate pipework scans (exceptions to this are noted):

"After a short (<1000 hours) period of rapid deposition near the source, ^{54}Mn migrates to the cold leg, with heaviest deposition in the lower temperature part of the heat exchanger and in the first part of the cold leg downstream of the heat exchanger. A small amount of ^{54}Mn circulates in the loop, based upon gamma scans of the unirradiated specimens. The count rate on the specimen immediately upstream of the four 316 SS irradiated specimens was 33 times lower than the counting rate on the first downstream specimen.

Cobalt-60 deposits preferentially in the loop hot leg, but with each succeeding gamma-scan significant ^{60}Co deposition is found further and further downstream. The cold leg deposition rate is less than one-tenth the hot leg deposition rate. There is some evidence that cobalt isotopes do not show a tendency for irreversible deposition in the minimum temperature locations where they deposit. The specimen just upstream of the four irradiated 316 SS had 9.1 times as much ^{60}Co activity as ^{54}Mn activity, despite the fact that the specific activity of the specimens was about three times higher in ^{54}Mn ."

Significant migration of ^{54}Mn into the hot trap was observed; it is not certain, however, whether the Zr at 1300°F in the hot trap provides an irreversible sink. Only slight amounts of ^{60}Co are present in the hot trap.

The deposition pattern of ^{54}Mn in the new economizer are quite similar to those observed in Figure 4. The ^{60}Co deposition is also given. (Note that the scales for ^{54}Mn and ^{60}Co differ by three orders of magnitude.)

Gamma-Scanning of STCL-1 During Loop Operation

The STCL-1 loop is gamma-scanned through the loop containment, as described in the previous quarterly report⁽²⁾. The results of the gamma-scanning are shown in Figures 5 through 10. The nuclide buildup shows basically the same trends as previously established. Note the rapid buildup of ^{54}Mn at Positions 24 and 30, the economizer outlet and the airblast cooler section, respectively. Also note the substantial buildup of both ^{54}Mn and ^{60}Co in the hot-trap economizer (Position 300) and the hot-trap body (Position 200).

Gamma-scanning through the STCL-1 containment will continue for the duration of Run 7.

STCL-2 and 3 Corrosion Testing

STCL-2 and 3 were run 3000 hours at 1120°F , 22 ft/sec, and 0.5 ppm oxygen. These loops operated with "nuclide traps" assembled in such a manner that segments of the traps could be weighed before and after testing. The loops have a second specimen chamber at the hot leg temperature just downstream from the first one. The second chamber provides a sodium velocity of 10 ft/sec. There is also a specimen chamber in the cold leg. A further description of the specimen chamber is given in Reference 9. Weight-change data for specimens in STCL-2 and 3 are shown in Table 8.

Expected Accomplishments Next Quarter

- The status and plans for STCL-1, 2, and 3 are shown in the Test Matrix, Table 9.
- Incremental analyses on 316 SS and other samples from STCL-2 and 3 will be completed.

HEDL 7602-57.21

FIGURE 4. ⁵⁴Mn and ⁶⁰Co Deposition in STCL-1 Economizer.

HEDL 7602-57.7

FIGURE 5. Nuclide Buildup in Hot Leg of STCL-1 (Position 15).

HEDL 7602-57.6

FIGURE 6. Nuclide Buildup in Center of Economizer in STCL-1 (Position 21).

HEDL 7602-57.9

FIGURE 7. Nuclide Buildup at Economizer Outlet in STCL-1 (Position 24).

HEDL 7602-57.11

FIGURE 8. Nuclide Buildup in Cooler of STCL-1.

HEDL 7602-57.10

FIGURE 9. Nuclide Buildup in Hot Trap Body (Position 300) in STCL-I.

HEDL 7602-57.8

FIGURE 10. Nuclide Buildup in Hot Trap Economizer (Position 300) in STCL-1.

TABLE 8
STCL-2 AND 3 WEIGHT-CHANGE DATA
3000 HOURS, 0.5 PPM OXYGEN

	Specimen grouping (number of specimens)	Temp. °F	Velocity ft/sec	ΔW $\mu\text{g}/\text{mm}^2$ average	Equivalent Corrosion Rates	
					mils/year	microns/year
<div style="display: flex; align-items: center;"> <div style="writing-mode: vertical-rl; transform: rotate(180deg); margin-right: 5px;">flow</div> <div style="border-left: 2px solid black; height: 100px; margin-left: 5px; position: relative;"> <div style="position: absolute; top: 0; left: -5px;">↓</div> </div> </div>	316SS Irradiated (4)	1120	22	5.3	0.076	1.9
	304SS-Chamber II (4) (first four in line)	1120	10	3.48	0.05	1.28
	304SS-Cold Leg (8)	800	2	+0.07(gain)	+0.01	+0.26
<div style="display: flex; align-items: center;"> <div style="writing-mode: vertical-rl; transform: rotate(180deg); margin-right: 5px;">flow</div> <div style="border-left: 2px solid black; height: 100px; margin-left: 5px; position: relative;"> <div style="position: absolute; top: 0; left: -5px;">↓</div> </div> </div>	316SS Irradiated (4)	1120	22	5.7	0.082	2.1
	304SS - II (4) (first four in line)	1120	10	4.1	0.059	1.5
	304SS-Cold Leg (8)	800	2	+1.08	+0.015	+0.40
<u>STCL-1, 3000 Hr Data*</u>						
	Irradiated 316SS	1120	22	4.7	0.068	1.7

* STCL-1 averaged data for 3000 hours, 0.5 ppm oxygen from Reference 4.

TABLE 9
TEST MATRIX FOR SOURCE TERM CONTROL LOOPS

<u>Loop</u>	<u>Specimen Temp/Oxygen</u> <u>minimun temp °F</u>	<u>ppm</u>	<u>Specimen Materials</u>	<u>Trap ?</u>	<u>Test</u> <u>Duration</u>	<u>Start/Finish</u>
STCL-1	1120 ⁰ /800 ⁰	<0.01 (hot trap)	I316,CW I316,SA U316,SA U316,CW	no	6000 h	3-75/12-75 Loop to be dismantled
STCL-2	1120 ⁰ /800 ⁰	0.5	I316,CW U316,CW U304,SA	yes	3000 h	8-75/12-76
STCL-3	1120 ⁰ /800 ⁰	2.5	I316,CW U316,CW U304,SA	yes	3000 h	8-75/12-75
proposed* STCL-2	920 ⁰ /600 ⁰	0.5	I316,CW U316,CW U304,SA	possibly	12000 h	1-76/7-77

*The proposed test at a 920°F specimen temperature will provide information on the magnitude of nuclide source rates at an FFTF core outlet of 900°F.

I - irradiated

U - unirradiated

SA - solution annealed

CW - 20% cold worked

- A more quantitative analysis of the STCL-1 gamma scanning data will be made.
- The present 3000 hour runs in STCL-1, 2, and 3 will be completed in December 1975.

Fission Product and Fuel Transport Behavior and Control - R. P. Colburn
and E. O. Badgett

Objective

The objectives of this program are (1) to determine fission product and fuel particulate transport kinetics and deposition patterns in LMFBF primary heat transport systems, (2) to determine the impact transport and deposition of these radionuclides will have on plant maintenance, and (3) to develop methods to monitor and control their deposition.

Summary of Accomplishments

- This program is a new subtask in the Radioactivity Control Technology Program, 189a FF057/GT-T. A literature search on the behavior of fission products and fuel in sodium revealed substantial differences in the observations and interpretation of data among the various investigators whose work was reviewed.
- Initial experimental investigation of the redistribution of ^{137}Cs in liquid sodium was performed in a thermal gradient capsule. The ^{137}Cs concentration in the cold end (255°F) of the capsule increased by a factor of 4 while the ^{137}Cs concentration at the hot end decreased by a factor of 4 relative to the uniform distribution observed when the capsule was isothermal at 1000°F.

Discussion

A literature survey completed this quarter revealed that there is general agreement among investigators with respect to qualitative trends of relative fission product release fractions from operating failed fuel pins. The volatile fission product species, ^{137}Cs and ^{131}I give large release fractions from failed pins where sodium has made contact with the

fuel while the less volatile, solid fission products and fuel show substantially smaller releases. However, there are substantial differences in the quantitative estimates of these release fractions and the subsequent transport behavior of the radionuclides in liquid metal systems. ^{137}Cs has been identified as the major radioactive contaminant in the primary coolant systems of LMFBR's which have operated with failed cladding.

Capsule tests were performed to determine the ^{137}Cs redistribution along a thermal gradient imposed on a sodium solution contaminating trace amounts of ^{137}Cs . The tests also provide radioactive specimens which are to be used for characterizing the nature of the deposits. The deposition behavior may provide information which will allow one to control the distribution in an LMFBR primary coolant system.

The capsule consists of a 3/4" diameter 304 SS tube, 17 inches long, containing a thermocouple well extending the length of the capsule. The capsule was filled with sodium in a glovebox and a 100 μCi source of ^{137}Cs deposited as CsNO_3 was placed near the top of the capsule. The capsule was heated uniformly to 1000 $^{\circ}\text{F}$ for 24 hours. This gave a relatively uniform distribution of ^{137}Cs as determined by gamma scanning the length of the capsule with a collimated NaI scintillation detector. Various temperature gradients were imposed on the capsule resulting in redistribution of the ^{137}Cs as shown in Figure 11. For a given gradient a steady-state Cs distribution was obtained within 24 hours. The temperature gradients were measured by moving a thermocouple along the length of the thermocouple well. The upper half of the capsule was held at 1000 $^{\circ}\text{F}$ while the lower end was cooled to achieve a given temperature distribution.

Conceptual design has begun on a capsule device to inject fission products and fuel particulates into sodium. This device will simulate release of fission products from an operating defected fuel pin during power transients.

Expected Accomplishments Next Quarter

- Start conceptual redesign of STCL-1 for fission product studies.

HEDL 7602-57.17

FIGURE 11. ¹³⁷Cs Distributions Along the Thermal Gradient Capsule.

- Complete the analysis of ^{137}Cs distribution data and characterize the nature of the deposits.
- Complete fabrication of the fuel/fission product injection device and begin transport studies in sodium capsules.
- Evaluate potential stand-ins for Pu for initial fuel/fission product behavior studies in sodium.
- Start fuel and fission product distribution studies in thermal convection loops.

Radioactivity Decontamination Process Development Program - R. P. Colburn, W. L. Kuhn, H. P. Maffei, and E. O. Badgett

Objective

The objective of this program is the development of decontamination processes for application to LMFBR primary heat transport system components which will permit contact maintenance and requalification of the component.

Summary of Accomplishments

- Activated corrosion product contamination of components in the hot (1050°F) region of the FFTF primary coolant loop is expected to occur by the mechanism of diffusion of radionuclides into the base metal. Subsequent decontamination will require removing the contaminated base metal to a depth achieving the desired decontamination factor. The requisite depth is estimated from a consideration of the simultaneous processes of diffusion, surface corrosion, and radioactive decay occurring during contamination. Based on recently estimated diffusion coefficients, a decontamination factor of 190 should be achieved by removing about 0.1 mil of base metal.
- The decontamination process development program was restructured as a joint effort between HEDL and Atomics International (AI) this quarter.

- Reagent screening tests under reflux conditions in the temperature range of 50⁰ to 100⁰C were performed using solution annealed and sensitized 304 SS specimens. These tests indicate that mixtures of acetic and formic acids are promising reagents for the decontamination process.
- A survey of the published literature was made covering the areas of LMFBFR decontamination experience and characterization of mass transfer deposits in sodium/stainless steel systems. The survey indicated previous LMFBFR decontamination processes will not be adequate to meet the projected requirements for FFTF or CRBRP. The survey also showed a good, general consensus concerning the character of mass transfer deposits in non-isothermal sodium/stainless steel systems.
- Incremental analysis of the surfaces from the STCL-1 economizer showed significant penetration of the ⁵⁴Mn into the base metal on both the tube and the shell.
- Electrochemical measurement equipment has been set up to continuously monitor the instantaneous corrosion rate of specimens undergoing candidate decontamination processes. The system has shown the large increase in corrosion rate of 304 SS in acetic/formic acid mixtures, when the solutions are heated to the boiling point, is associated with a rapid change in the specimens electrode potential (relative to a chromel electrode). These preliminary measurements indicate depassivation occurs at the boiling point under refluxing conditions and that the phenomena may also be affected by aeration of the solution.

Discussion

Modeling

Activated corrosion products are expected to transport to FFTF primary coolant hot-leg loop components primarily in the zero valence state. Consequently, the various radionuclides are readily available for diffusion into the base metal of the components, so that the contamination "alloys" with

the base metal. Subsequent decontamination thus requires the removal of the base metal to a depth which achieves the desired decontamination factor.

This section describes how the depth of metal removal required to achieve a given decontamination factor is estimated. The estimate is based on the assumption of ordinary (i.e., not grain boundary) diffusion of radionuclides into the metal during the simultaneous decay of the radionuclides and corrosion of the metal surface. A decontamination factor is obtained from the concentration profile of a radionuclide with depth into the metal. The concentration profile is obtained by solving a partial differential equation describing the unsteady-state diffusion, surface corrosion, and decay occurring for a generalized boundary condition.

The greatest radionuclide penetration is reached asymptotically at long times; this presents the worst case with respect to decontamination requirements and is the case considered here. From the unsteady-state concentration profile one finds that a sufficient condition for reaching the maximum penetration is $\lambda t = 9$ for a given radionuclide. The decontamination factor attained for a given depth of metal removed is then found to be

$$\text{D.F.} = \exp (x / x^*)$$

$$\text{where } x^* = \sqrt{\frac{D}{\lambda}} / \left(\sqrt{1 + u^{*2}} + u^* \right),$$

$$\text{and } u^* = u / \sqrt{4D\lambda}.$$

Here D is the diffusivity (cm^2/sec),

u is the surface corrosion rate (cm/sec), and

λ is the decay constant for the radionuclide considered ($1/\text{sec}$).

For the FFTF primary coolant loop hot leg u^* is expected to be small; hence we assume

$$x^* = \sqrt{\frac{D}{\lambda}}.$$

A recent estimate of D for ^{54}Mn in 316 SS at 1050°F (FFTF hot leg temperature) is $6.23 \times 10^{-17} \text{ cm}^2/\text{sec}$, obtained from the literature⁽¹⁰⁾ and RCT sodium corrosion experiments. Using $\lambda = 2.56 \times 10^{-8}/\text{sec}$ for ^{54}Mn , we estimate

$$x^* = \sqrt{\frac{6 \times 10^{-17}}{2.56 \times 10^{-8}}} = 4.84 \times 10^{-5} \text{ cm} = 0.019 \text{ mil.}$$

Therefore, removing 0.1 mil of base metal should achieve a D.F. of 190.

There is considerable uncertainty in this estimate, for the following reasons:

- (1) Grain boundary diffusion (not considered above) may be substantial, thus greatly deepening the steady-state concentration profile and consequently reducing decontamination factors.
- (2) The contaminated surface may be very rough, requiring more metal to be removed overall.
- (3) The diffusion coefficient is still not well characterized.

Program Structure

The decontamination process development program was restructured to incorporate process development work at Atomics International (AI). HEDL has lead responsibility for the program with the ultimate objective to demonstrate processes in a large decontamination facility. In support of this objective HEDL will (1) continue surface characterization of radioactive mass transfer deposits from the STCL facilities, (2) evaluate promising candidate decon processes on various test feature configurations, (3) perform analysis to define the nature and magnitude of the contamination on FFTF components for their projected operation, and (4) design, fabricate, and operate a large-scale decontamination facility.

AI will continue reagent screening tests followed by process development and process optimization to define the candidate decontamination processes for LMFBR components. Decontamination of components for reuse in the reactor and for disposal will be addressed.

Screening Tests

Reagent screening tests were performed under refluxing conditions in the temperature range from 50°C to 100°C using solution annealed and sensitized 304 SS specimens. The apparatus is shown in Figure 12.

The reagents, test temperatures, exposure times, weight losses (reported as uniform metal removal), and metallurgical observations are listed in Table 10. Most of the reagent combinations tested below 100°C gave insufficient metal removal. Several reagents produced significantly higher metal removal rates at 100°C but generally metal removal was extremely non-uniform or the samples were attacked intergranularly as shown in Figures 13, 14, 15, and 16. The 5% Turco 4521 solution was extremely aggressive on the cold worked portions of the 304 SS samples. The two promising processes are a mixture of 30% acetic/5% formic acid at 100°C and 1M oxalic acid/0.5M ammonium citrate at 100°C. The two processes resulted in uniform metal removal rates of approximately 1 mil in 10 days for the acetic/formic mixture and 0.4 mil in 10 days for the oxalic acid/ammonium citrate mixture. A plot of metal loss versus time for the former solution is shown in Figure 17. The metallurgical condition of samples exposed to these two solutions is shown in Figures 18, 19, and 20.

Electrochemical measurements (linear polarizations) were used to monitor the corrosion rate of 304 SS in 30% acetic/5% formic acid solution. With this technique the specimen acts as a "working" electrode and the corrosion rate of the specimen (rate of oxidation) may be evaluated from an analysis of the behavior of the applied current resulting from polarization by a few millivolts on either side of the natural potential (relative to a chromel electrode) of the test specimens^(11,12). The apparatus is shown in Figure 21. Preliminary measurements indicate the passivated stainless steel surfaces are rapidly activated in the boiling solution; the specimen potential shifts from +.20 to -.42 volts and the corrosion rate increases by over 2 orders of magnitude. Spontaneous repassivation of the surface occurs upon cooling to room temperature but the rate of repassivation appears to depend on aeration of the solution.

TABLE 10

METAL REMOVAL BY VARIOUS CANDIDATE DECONTAMINATION SOLUTIONS

Solution Composition	Solution Temperature	Exposure Time	Metal Removal Calculated from Weight Loss	Maximum Non-uniform Attack Measured by Metallography		Remarks
				Pitting	Intergranular	
30% Acetic Acid	100°C	288 hrs	1.1 mils	0.5 mils	0	
5% Formic Acid	100°C	192 hrs	0.6 mils	0.5 mils	0	
	75°C	288 hrs	0.1 mils	0	0	No weight loss after 48 hours
	50°C	288 hrs	<0.01 mils	---	---	
0.1M Oxalic Acid	100°C	192 hrs	0.4 mils	0.7 mils	0	No weight loss after 48 hours
0.5M Ammonium Citrate	75°C	192 hrs	<0.01 mils	0	0	
	50°C	192 hrs	<0.01 mils	---	---	
0.1M Oxalic Acid	100°C	168 hrs	0.1 mils	---	---	
0.1M Sodium Fluoride	75°C	168 hrs	0.03 mils	---	---	No weight loss after 120 hours
0.1M Aluminum Nitrate	50°C	168 hrs	0.01 mils	---	---	No weight loss after 120 hours
0.5M Oxalic Acid	100°C	192 hrs	0.3 mils	1 mil	2 mils	
0.1M Sodium Fluoride	75°C	192 hrs	0.2 mils	1 mil	3 mils	
0.1M Aluminum Nitrate	50°C	192 hrs	0.07 mils	---	---	
5% Nitric Acid	100°C	96 hrs	0	---	---	No weight loss in 96 hours
	75°C	96 hrs	0	---	---	No weight loss in 96 hours
	50°C	96 hrs	0	---	---	No weight loss in 96 hours
5% Nitric Acid	100°C	192 hrs	0.5 mils	0.5 mils	4 mils	
1% Chromic Acid	75°C	192 hrs	0.02 mils	0	1 mil	
	50°C	192 hrs	<0.01 mils	0	0	
5% Turco 4521	100°C	192 hrs	1.7 mils	10 mils	0	All pitting associated with cold worked metal
	75°C	216 hrs	0.01 mils	---	---	
Sulfurous Acid	100°C	48 hrs	1.1 mils	3 mils	0	Solution saturated with SO ₂
	95°C	96 hrs	<0.01 mils	---	---	Solution saturated with SO ₂
	25°C	48 hrs	<0.01 mils	---	---	Solution saturated with SO ₂

FIGURE 12. Apparatus for Performing Screening Tests Under Refluxing Conditions. (Photo No. 75-3243-1cn)

FIGURE 13. Reaction of 1% Chromic Acid 5% HNO₃ on 304 SS for 192 Hours at 100°C.

HEDL 7510-205.6

FIGURE 14. Reaction of Sulfurous Acid on 304 SS for 48 Hours at 100°C.

(12X)

(12X)

(50X)

(50X)

FIGURE 15. Reaction of 5% TURCO 4521 Solution on 304 SS for 192 Hours at 100°C.

HEDL 7510-205.3

HEDL 7510-205.4

FIGURE 16. Reaction of 0.5M Oxalic Acid, 0.1M Sodium Fluoride, 0.1M Aluminum Nitrate on 304 SS.

HEDL 7602-57.19

FIGURE 17. Metal Removal by 30% Acetic/5% Formic Acid Decontamination Solution.

250 HOUR EXPOSURE AT 100°C

250 HOUR EXPOSURE AT 75°C

HEDL 7510-205.1

FIGURE 18. Reaction of 30% Acetic Acid 5% Formic Acid on 304 SS.

HEDL 7510-205.5

FIGURE 19. Reaction of 30% Acetic Acid/5% Formic Acid on 304 SS for 192 Hours at 100°C.

192 HOUR EXPOSURE AT 100°C

192 HOUR EXPOSURE AT 75°C

REACTION OF 0.1M OXALIC ACID. 0.5M AMMONIUM CITRATE ON 304 SS

HEDL 7510-205.2

FIGURE 20. Reaction of 0.1M Oxalic Acid, 0.5M Ammonium Citrate on 304 SS.

FIGURE 21. Apparatus for Conducting Electro-Chemical Corrosion - Rate Measurements. (Photo No. 756473-1)

Surface Characterization

Incremental analyses of surfaces from the STCL-1 economizer have been completed. Radiochemical profiles contrasted with the elemental composition profiles reported last quarter⁽²⁾ in which the deposit was confined to a layer less than 1 μ m thick and relatively uniform base metal compositions were found below it. The ^{54}Mn profile extended far into the base metal through the entire depth sampled by incremental polishing. Up to 10% of the original ^{54}Mn activity remained in the specimens after removing 7 μ m from the surface. A typical ^{54}Mn penetration profile from the heat transfer surface near the midpoint in the economizer is shown in Figure 22.

Expected Accomplishments Next Quarter

- A program plan will be prepared giving the work scope and test matrix for the joint HEDL-AI decontamination program.
- Promising candidate processes from the screening tests matrix will be evaluated by electrochemical measurements.
- Feature test assemblies consisting of ASTM tensile specimens in shroud sleeves will provide test pieces in tension with crevice and threaded regions. These specimens will be sensitized in sodium. A reference sodium removal process will be followed by decontamination processes selected from the screening test matrix.
- The metallurgical effects of selected decontamination processes on 316 SS, hard facing materials, and weld regions will be evaluated.
- Small glass/Teflon decontamination test loops will be erected to evaluate selected decontamination processes using radioactive specimens from STCL-1.
- Surface characterization of STCL-1 surfaces will continue with emphasis on determining the surface morphology of the deposits in the STCL-1 economizer.

HEDL 7602-57.18

FIGURE 22. ^{54}Mn Penetration Profile of STCL-1 Economizer Shell, Specimen E-3-D-P (1020°F).

Radionuclide Trap Development - J. C. McGuire and J. L. Bevan

Objective

One area of the Radioactivity Control Technology (RCT) research effort is the development of trapping materials and trap designs suitable for the removal of long-lived gamma emitting radionuclides from the primary loop sodium of the FFTF/LMFBR.

Summary of Accomplishments

- A second set of prototype nuclide traps has completed 3000 hours of test in Source Term Control Loops 2 and 3. These were scale versions of trap designs suitable for use in an LMFBR. Both traps were effective in reducing downstream activity. A third design is now in test.

Discussion

The effort to design and fabricate an effective nuclide trap for service in an LMFBR requires a large amount of background data. This effort is now directed toward a nuclide trap test in EBR-II using the best available design and trap materials. The necessary data include predicted corrosion rates, activity deposition with varying geometry, most effective trap materials, and flow parameters of suggested designs.

The work done during the past period was all aimed at furnishing these necessary data.

Expected Accomplishments Next Quarter

- Gamma scanning will continue on the STCL-2 and 3 nuclide traps together with associated cold zone locations, to observe any possible changes in activity buildup and to see how efficiently the traps perform.
- Further preparations will be made for the formal documentation of a request for Approval-in-Principle necessary to place a prototype nuclide trap in EBR-II.

Tritium Control Program - J. C. McGuire, W. L. Kuhn, and E. O. Badgett

A tritium control program was formulated consisting of two areas: (1) model development for tritium transport in an LMFBF and (2) an experimental program to measure hydrogen/deuterium/tritium diffusion rates through 304 stainless steel and Croloy with surfaces prototypic of those for an LMFBF.

Summary of Accomplishments

Modeling

- The initial thrust of the modeling subtask will be to update the previous estimates^(13, 14) of the tritium release expected during operation of the FFTF. This will be pursued via a National Tritium Program plan being prepared by Argonne National Laboratory (ANL) and HEDL. Tritium release models developed at ANL-E, ANL-W, and HEDL will be modified to conform to a uniform tritium transport model, based on the most recent ANL-E model⁽¹³⁾ and updated as experimental data become available.

Experimental

- Two different diffusers fabricated of 304L stainless steel have been used in determining the comparative diffusion rates for "clean" (as received, washed, and outgassed) and "oxidized" stainless steel where the oxide is the result of high temperature exposure to a low partial pressure of dry oxygen. Contrary to much of the published literature, this form of oxide surface was not found to inhibit hydrogen diffusion. On the other hand, a diffuser inadvertently oxidized slightly by a small amount of water vapor in test hydrogen was found to reduce diffusion rates by an order of magnitude. A subsequent high temperature heat treatment of this diffuser under hydrogen yielded a 4-fold increase in diffusion rates.
- A comparison of H₂ and D₂ diffusion rates shows reasonable adherence to the $(\sqrt{M})^{-1}$ diffusion law.

Discussion

Introduction

The broad goal of the tritium diffusion program is a determination of the extent of the problem presented by tritium created within the reactor and the development of methods suitable for control or containment of that tritium.

Two problems must be addressed: the control of tritium and the control of hydrogen. Tritium is created by ternary fission within the fuel and even more frequently as a byproduct of the neutron boron reaction within the control rods. Although present estimates indicate a total product of less than 50 cc of tritium per day, the ready accessibility of hot containment walls and the significant diffusion rate through those walls at reactor temperatures could possibly cause some concern for levels released to the environment.

The second problem is the evolution of hydrogen at the secondary sodium/steam line interface. Hydrogen generated in this area can diffuse through the Croloy tubing to the secondary sodium. From there it will be carried to the primary heat exchanger where it can diffuse into the primary sodium stream. A small amount of this hydrogen is actually beneficial because it will promote deposition of tritium in the primary and secondary loop cold traps, but too much hydrogen could have a deleterious effect on the fast reactor core nucleonics.

Modeling

A tritium program coordination meeting was convened by ANL at EBR-II on August 12 as a prerequisite for writing a program plan. It was informally agreed that the most recent ANL tritium transport model⁽¹³⁾ is an adequate basis for the refinement of a common ANL/HEDL model. This model will be used by HEDL to estimate the release of tritium expected during operation of the FFTF, as a function of plant operating variables. The role of cold trapping and hydrogen ingress is to be emphasized.

An improved knowledge of hydrogen-tritium interaction in a sodium cold trap and the hydrogen generation, ingress, and permeation rates throughout

LMFBR materials are required before the tritium transport model can be refined. This information will be forthcoming from experimental tasks to be defined as part of the National Tritium Program.

It is the consensus of previous workers that an oxide coating on stainless steel can provide a drastic reduction in diffusion rate. Reductions as great as three orders of magnitude have been reported; however, some workers have reported little or no effect. In an effort to resolve these anomalous results, it will be necessary to form the oxide coating in a carefully characterized manner. Two variables appear to play a part: (1) temperature and (2) the method used for oxidation. For FFTF/LMFBR applications the temperature at which any indigenous oxide layer would be applied would not exceed 600°C and would probably be formed at temperatures up to 550°C . Therefore, the temperature for experimental application is confined to 550°C .

There appear to be two methods for application of the oxide layer: (1) directly from surface reaction with oxygen or (2) from the reaction of water vapor, releasing H_2 . If these two methods can be used separately, on separate diffusers, the effectiveness of each method can be evaluated. Once the most effective method is determined, the protective gas barrier around primary containment (now 99% N_2 /1% O_2) can be adjusted to provide the proper oxide coat for maximum diffusion inhibition.

Experimental

The first diffuser, described in Reference 2, has been evaluated in a series of shakedown tests of the diffusion equipment. Using this diffuser, permeation rates close to literature predictions were obtained for H_2 . During the course of these tests at 600, 500, and 400°C , a sharp drop in the diffusion rate was observed. Investigation showed that the H_2 contained ~ 10 ppm H_2O .

The source of H_2 was then purified by running the incoming line to the diffuser through a liquid N_2 cold trap. As soon as this precaution was adopted the degradation in diffusion rates stopped; however, the rate had already been lowered by a factor of 16. Subsequent heat treatment of the diffuser in H_2 at temperatures up to 750°C had no significant effect on the

diffusion rates; but a sharp excursion to 900°C for 30 minutes increased the rates by a factor of four.

With the knowledge gained in the first diffuser runs, a second diffuser (No. 2) was fabricated featuring a new design. As seen in Figure 23, this diffuser is fitted to the glass diffuser tube by means of a ground glass joint. The diffuser tube is all fused silica with graded seals to the remainder of the equipment. Using this design the diffuser can be changed by cutting only one line, the connection to the capacitance manometer, and opening the ground joint.

Diffuser No. 2 was carefully outgassed in increments of 50°C to 550°C. Diffusion rates for H₂ and D₂ at 350, 400, and 450°C between 0.5 and 12 Torr are seen in Figure 24. The diffusion rates for D₂ are lower than those of H₂ by the predicted factor of $(\sqrt{2})^{-1}$ at the various temperatures and pressures.

Oxidations were all carried out with 99% N₂/1% O₂, the planned blanket gas between the primary containment and the reactor vessel. The gas from the cylinder was passed through a cold trap suspended just over the surface of liquid nitrogen. The purpose in mind was to achieve the lowest possible dew point without condensing O₂. This gas was then fed to the diffuser at atmospheric pressure and bled out through a fine capillary tube in water to observe positive flow.

In total, three oxidation runs were performed at 550°C for 15 minutes, 1 hour, and 1 hour, respectively. After each oxidation, diffusion rate curves were determined at 500, 450, 400, and 350°C from 10 Torr to as low as 0.06 Torr ΔP. Data are plotted in Figure 25. In spite of preconceived notions to the contrary, the diffusion rate remained essentially unchanged from the values recorded for bare metal. Note that the data become more scattered as the temperature decreases. This observation indicates that dry 99% H₂/1% O₂ does not provide a protective oxide coating when treated in contact with 304L stainless steel at reactor operating temperatures.

To determine whether the stable or protective form of oxide might be a hydrate, the No. 2 diffuser was exposed to a humid atmosphere overnight at room temperature. However, the diffusion rates were found to be unchanged.

HEDL 7602-57.13

FIGURE 23. Improved Diffuser No. 2.

HEDL 7602-57.15

FIGURE 24. Diffusion Rates For Hydrogen and Deuterium Before Oxidation - Diffuser No. 2.

FIGURE 25. Observed Diffusion Rates For Diffuser No. 2.

Further oxidation runs using a moist oxidizing atmosphere will be tried next quarter on a new diffuser tube.

Expected Accomplishments Next Quarter

- Diffuser No. 3 will be fabricated and tested. Tests will include flow characterization as installed, careful firing under dry hydrogen @ 850-900°C with a re-determination of diffusion rates, and a progressive oxidation in argon using increasing H₂O contents.
- Analyses of Diffusers 1 and 2 by SEM, Auger, and metallographic examination will be made to determine differences in surface composition and characteristics.
- A fourth diffuser will be tested. Tests will include "water vapor" oxidation and subsequent "dry oxygen" oxidation to determine whether the water-vapor-created protective coating is stable.
- Laboratory equipment and monitoring instrumentation will be readied for conducting experiments with tritium.

SODIUM SYSTEMS DEVELOPMENT

J. L. Ballif

Sodium Removal - G. B. Barton, M. D. Crippen, and C. W. Funk

The objective of this task is to provide sodium removal technology for the support of FFTF and HEDL operations. This includes methods, documentation, and consultation for design and operation of large sodium removal facilities required at this site.

Summary of Accomplishments

- Completed a study of sensitization temperature versus caustic stress corrosion cracking.
- Qualified the AMCF hydrogen meter for use.
- Completed two additional blind temperature runs in SRDA-I.
- Successfully removed sodium and frost from CRCTA viewport.
- Activated the SRDA-II, a new 2.5 ft. diameter x 12 ft. high facility, that uses alcohol as the sodium removal media.
- Successfully completed Phase II sodium removal from IHX model parts.

Discussion

Components submitted to the sodium removal process (Water Vapor Nitrogen-WVN) can exhibit many degrees of sensitization. In light of this, specimens were prepared and examined to determine if argon-exposed sensitization would affect the susceptibility to caustic stress corrosion cracking (SCC) in Type 304 SS. Three representative conditions were tested: (1) as manufactured, (2) heat treated in argon at 1050⁰F for 48 hours, and (3) heat treated in argon at 1250⁰F for 48 hours. Duplicate specimens (sleeved, stressed to 100% of yield strength) of each were exposed to 50 wt% NaOH at 300⁰F for 14 days. (These conditions are known to crack by design.) Examination of the data seen in Table 11 disclosed a range of

depths for the deepest visible crack of 3.0 to 8.9 mils. These are worst-case cracking conditions where cracking is to be expected. The tests could not differentiate between the three heat treatments; hence, it was concluded that caustic SCC on 304 SS is independent of the degree of sensitization at 1250⁰F, 1050⁰F, or in the "as manufactured" conditions.

The gas chromatograph purchased to measure the hydrogen evolved in the Alkali Metal Cleaning Facility (AMCF) has successfully been qualified for use by more than 1000 hours of successful operation. It displays the requisite sensitivity and rapidity of response specified.

Two additional blind runs, runs in which the temperature parameter was not visible to the operator, were successfully completed. In run 117-12 three pots containing sodium (4, 8, and 12 in. in diameter) were placed at 0, 2, and 8 ft. respectively in the 12 ft. high x 2.5 ft. diameter vessel. In addition, each pot was covered with a shroud which only allowed access by the gas through four 1/2 in. diameter holes. No instabilities were seen. The pot covers slowed down the reaction time thus extending the elapsed time, but the process was sensitive to reaction and allowed precise control.

The second run, No. 188, was in the small SRDA vessel. It was likewise successful, although the run was impaired by equipment problems which allowed a momentary over-temperature. The results are considered very encouraging.

A 6-in. diameter sodium valve equipped with a glass viewport was processed for cleaning. This viewport was used for some time on the CRCTA vessel to observe the sodium pool. After several hundred hours of operation the viewport fogged and the actuating mechanism jammed from the pickup of the frost. This valve was successfully cleaned and made operative by use of the WVN process. Processing time was 142 hours in the reaction phase at 5% water vapor and 26 hours for rinse--for a total of 168 hours.

The first shakedown run was completed successfully in the Sodium Removal Development Apparatus-II (SRDA-II). The purpose of the run was to check the system operability of the alcohol cleaning (ROH) facility using

TABLE 11
EFFECT OF SENSITIZATION IN CAUSTIC STRESS CORROSION CRACK DEPTHS
CONDITION: SLEEVED, STRESSED SPECIMENS EXPOSED
TO 50 WT% NaOH FOR 14 DAYS @ 300°F

<u>Sensitization Treatment in Argon</u>	<u>Depth of Maximum Crack (mils)</u>	<u>Specimen # & Right or Left Specimen Edge</u>
As Machined	8.5	A-R
	3.0	A-L
	8.5	B-R
	3.5	B-L
48 hr 1050°F	6.0	A-R
	4.1	A-L
	8.9	B-R
	5.7	B-L
48 hr 1250°F	3.5	A-R
	4.6	A-L
	5.7	B-R
	6.3	B-L

Conclusion: Depth of SCC is not affected by sensitization

a real piece of hardware (a 6-in. diameter freeze seal valve obtained from LMEC). The operation of the facility was generally satisfactory with only minor problems. The alcohol removed about 4.5 lbs. of sodium from the valve in 10 days. Post process examination of the valve noted some residual sodium remaining. The commissioning of this facility now provides the capability of cleaning components that can be fit into a space 2.5 ft. diameter by 12 ft. high with either WVN or ROH.

Several models of the IHX were fabricated by Foster-Wheeler, the FFTF IHX vendor, for sodium removal testing. These subscale models were plant scale except for length and number of tubes. After WVN processing of one model three times at HEDL, it was sectioned metallographically looking for possible caustic stress corrosion cracking. Possible sites were 1) locking pins welds, 2) tubesheet-to-tube welds, 3) tube-support plates because of possible limited access of reagent. Examination of those areas revealed artifacts such as delamination, duplex grain structure, surface defects in heat-affected zones of welded areas, deep grain-boundary etching, and some intergranular attack. None of these artifacts are characteristic of caustic stress corrosion. It was tentatively concluded that these were non-process (WVN) oriented artifacts which were probably caused by the fabrication process. Pieces of an unprocessed IHX model were split into two pieces and then wet in sodium. One was processed by vacuum distillation and one by WVN to remove the sodium. Thorough metallographic examination of both specimens showed that both structures were identical and no artifacts were seen in either. This strongly supports the conclusion that fabrication processes, not the WVN treatment, affected the first IHX.

Expected Accomplishments Next Quarter

- The AMCF will be qualified.
- The IVHM will be cleaned of sodium for examination.
- The IHX model processing will be completed.

SODIUM SYSTEMS ENGINEERING

D. W. Bennett

FFTF System 85 Procurement - D. W. Bennett and J. A. Yount

Objective

Sodium Technology has provided specifications and design drawings for procurement of System 85 Sodium Characterization System (SCS) components and is actively providing technical supervision of fixed price procurement contracts. The present objective is to complete procurement of the SCS components and associated control panels and then participate in installation testing and startup of the system.

Summary of Accomplishments

Milestones achieved this quarter include:

- Piping and pressure boundary fabrication was 50% completed.
- The SCS packaging and shipping plan was approved.
- Small sodium valves were delivered to the system fabricator.
- Vendor design of the Sodium Sampling Cell Crane was completed and approved.
- The System 85 Operation and Maintenance Manual reached 50% completion.

Discussion

Major piping subassemblies between valves were completed and valve installation has begun. The piping, with valves installed, will then be set into the previously constructed support frames. Work flow at the vendor's plant is proceeding satisfactorily under code-required controlled conditions. A HEDL quality assurance audit revealed general compliance with the quality assurance plan.

Fabrication of the System 85 control panels is proceeding on schedule at another fixed price contractor's plant. The control panels and the mechanical equipment will be delivered to the FFTF site in December 1975 for installation.

The System 85 in-cell crane design has been completed. Potential interferences with the crane in the cell have been evaluated and necessary design modifications have been made.

The System 85 Operations and Maintenance Manual reached 50% completion. General considerations for safe operation of the Multipurpose Samplers and Plugging Temperature Indicators have been drafted. Vendor-supplied information packages have been examined and specific requirements for operation of these vendor-supplied components have been inserted in the Manual. Completion of the Manual is contingent on vendor information packages that have not yet been provided.

Expected Accomplishments Next Quarter

- Pressure boundary fabrication of the SCS will be completed.
- SCS control panels will be delivered.

Cold Trap Testing - G. R. Bloom, K. B. Sowa, T. J. Owen, W. L. Kuhn, and D. W. Bennett

Objective

The objective is to provide technical assistance for design, procurement and operation of the FFTF and test facility cold traps.

Summary of Accomplishments

- A test request for evaluation of a modified closed loop cold trap was received and fabrication of the test fixture was initiated.
- The vendor-supplied revised thermal hydraulics design report has been evaluated.

Discussion

FFTF cold traps have been designed and fabricated, and lead units have been provided for design verification testing. Four cold trap configurations have been provided as part of this contract. They are:

Type I - 60 gpm NaK cooled for FFTF primary HTS

Type II - 15 gpm gas cooled for FFTF secondary HTS

Type CT - 3 gpm gas cooled for FFTF primary and secondary closed loops

Type III - 60 gpm NaK cooled for HTSF (same crystallizer as for Type I but different economizer)

Testing of the Type III and Type I Cold Traps on HTSF has shown that serious thermal performance deficiencies exist in the economizer sections of each. HEDL has modified the original plan to test all three FFTF cold traps as provided as a result of the Type III and Type I testing. The original plan called for immediate testing of the Type II cold trap at LMEC and for testing the Type CT trap at HEDL. Because deficiencies in performance of the Types I and III economizers can be readily extrapolated to the Type CT and to the Type II units, testing of these cold traps had been put on hold until a fix or a replacement can be determined for each.

The preliminary design for the replacement closed-loop cold trap economizer was transmitted to the contractor for information in preparing the final design. This final design is very similar to the preliminary design that was proposed by Sodium Technology with a helical single tube-in-shell arrangement. However, the vendor does not provide a method to maintain concentricity of the inner tube within the shell. The preliminary design accomplished by Sodium Technology featured a helical, wire wrap as a spacer between the shell and tube to maintain concentricity. Eccentricity between the tube and shell will result in some flow maldistribution and in decreased heat transfer performance. The vendor has accounted for this situation by increasing the active length.

Testing of the Type II cold trap has recently been completed at LMEC. Crystallizer performance experienced during these tests was used to modify the heat transfer model for the closed-loop crystallizer. As a result, the uncertainty of the closed-loop crystallizer performance has been substantially reduced. Sufficient uncertainty with regard to the performance of the new closed-loop cold trap economizer exists to warrant testing in liquid sodium.

A test request for the closed loop cold trap has been received from FFTF Engineering. The vendor will supply a closed-loop trap with a new helical economizer for testing in the HEDL Transient Test Loop (TTL). A test fixture is presently being fabricated at HEDL which will allow continued use of the cold trap on TTL after testing is completed. Provision is made for attachment of a sodium characterization system at a later date.

Expected Accomplishments Next Quarter

- Delivery of the test cold trap is expected in December, 1975. The test fixture will be completed and ready for installation of the cold trap. A test procedure will be prepared and approved.

CRBRP System 85 - J. A. Yount and D. W. Bennett

Objective

Provide specifications for the CRBRP Sodium Characterization System complete with attendant drawings, stress analysis, and design report.

Summary of Accomplishments

- Preliminary design was completed.
- A formal preliminary design review was convened.

Discussion

The preliminary design for CRBRP System 85 was completed and issued for comment. A design review meeting was hosted at Atomics International (AI) in which the preliminary design was evaluated for conformance with the System Design Description.

Changes to CRBRP SDD 85, both mechanical and instrumentation, were discussed. HEDL presented conceptual CRBRP sodium characterization system components. Based on the resulting discussion, further studies will include:

1. An evaluation of the need for a one-inch bypass flow control valve on the distribution loops.
2. Justification for provision of two multipurpose samplers for the primary heat transport system.
3. Analysis of lifting lug design to conform to CRBRP plant requirements.
4. Reevaluation of the need for and design of module drip pans.
5. A trade-off study of the relative costs of incorporating PTI and MPS instrumentation in the same panel on each system, versus separate panels for each on the primary system and Ex-Vessel Storage Tank.

HEDL concepts were generally well received, indicating that HEDL's recommendation for redesign of connecting piping as opposed to direct use of FFTF components is well justified.

The HEDL recommendation to install an oxygen-hydrogen meter in the HTS primary system was discussed. Generally, the CRBR community is in favor of the in-line meter concept but is not fully convinced that the ceramic oxygen meter probe exhibits long enough mean-time-between-failures to be useable. HEDL cited recent experience with the oxygen meter locally and design improvements that are underway involving the use of a ceramic button rather than the fragile tubular probe.

It was determined that action on the HEDL recommendation would be postponed until the new meter concept has been evaluated.

Expected Accomplishments Next Quarter

- A detailed hydraulic analysis will be provided for the system.
- Stress analysis will be initiated.
- The number of primary heat transfer system multipurpose samplers will be determined.

Freeze Vent Testing - G. R. Bloom, D. W. Bennett, and A. Deichsel

Objective

Provide procedures for FFTF vacuum and gas displacement sodium fill of the FFTF and verification that the freeze vents will prevent liquid sodium from entering the inert gas system in FFTF.

Summary of Accomplishments

- Gas displacement sodium-fill tests were completed.
- Vacuum-fill tests were completed.
- Completed freeze vent standoff-pipe tests.

Discussion

The FFTF Freeze Vent Test was performed on the Transient Test Loop in 336 Building. The gas-displacement fill, vacuum fill, plug stability, and the freeze vent standoff tests were completed.

Gas displacement fill tests were performed at sodium fill rates between 1 and 11 gallons of sodium per minute. The sodium gas interface level was 11.3 inches below the center of the freeze vent outlet tee (-11.3 inches) after the simulated 1 gallon per minute gas displacement fill test. Six tests at sodium-fill rates between 4 and 6.5 gallons per minute resulted in freeze vent plug positions between -7 and -11 inches. The average sodium-fill rate and plug position for the six tests were 5 gallons per minute and -9.5 inches, respectively. Three tests were performed at an average fill rate of 10.6 gallons sodium per minute which resulted in an average plug height of -0.8 inches. A small amount of sodium actually went through the freeze vent on two of the three high flow rate tests but a solid plug did form in each test. These tests demonstrate that a nominal sodium-fill rate of 5 gallons per minute would result in a sodium plug about midway up the freeze vent. Sodium-fill rates of 10 gallons per minute under gas displacement conditions will result in some sodium passing through the freeze vent before a solid plug is formed.

Six vacuum-fill tests were performed with the freeze vent at sodium-fill rates between 6.5 and 31 gallons per minute. The sodium plug position varied between -14 inches for the low sodium fill rates and -12 inches for the high sodium-fill rates. The tests demonstrate that the freeze vent will perform properly under vacuum fill conditions at sodium-fill rates up to 31 gallons per minute.

Two freeze vent plug stability tests were performed on sodium plugs formed under gas-displacement fill conditions. A 1 gallon per minute fill plug was tested for stability by heating the freeze vent standoff to 600°F with the ambient-air temperature at 150°F. Once steady state conditions were obtained, the sodium pressure was increased to 200 psig. Sodium plug position measurements before and after the test demonstrated that the plug moved from -11.3 inches to -7 inches during the test. A similar test with a plug originally at -10 inches formed by a 4 gallon per minute gas displacement fill resulted in a plug migration to -7 inches. The temperature at the base of the freeze vent was 516°F for the first stability test and 525°F for the second stability test.

The freeze vent standoff temperature profile test was performed with 1050°F sodium flowing through the 2 inch test section at 60 gallons per minute. The sodium pressure upstream of two valves was maintained at about 200 psig for 1 1/2 hours during the test. The temperature profile for the standoff is given in Figure 26. Comparison of the freeze vent sodium plug level before and after the standoff test demonstrated that the sodium plug moved from -7 inches to -2 inches during the test.

The sodium plug which moved during the standoff test was the same plug which was apparently stable during one of the stability tests. The plug moved during the standoff test under conditions of lower temperature and pressure than the test temperature and pressure applied during the stability test. The high sodium pressure was applied to the freeze vent for 1 1/2 hours during the standoff test and for 1 hour during the stability test.

FIGURE 26. Standoff Test Temperature Profile.

HEDL 7602-57.12

If plug movement were pressure-time dependent, then the stability tests may not have allowed enough time for plug movement and the plug position was still changing when the stability tests were terminated.

Expected Accomplishments Next Quarter

- Additional freeze vent testing is planned in order to determine the final stable freeze vent sodium plug position which can be expected for FFTF operating conditions.

SODIUM SYSTEMS ANALYSIS

J. J. McCown

Fuel Failure Monitoring (FFM)

Objectives

Develop, design, demonstrate and procure subsystems and components for FFTF and CRBR System 94, Fuel Failure Monitoring. Establish a technical base which will provide test data and assistance to reactor operations.

Summary of Accomplishments

- A majority of the documentation required prior to fabrication of FFM equipment was completed. Fabrication work is underway and all procurements are on schedule. Most of the components required for upgrading an existing spectrometer-computer for use on FFTF have been obtained, installed, and are under test. Design work was finished on an automatic liquid nitrogen supply system.
- The Cover Gas Monitoring Proof test on EBR-II continued to analyze fission gas releases. Improvements in the system decreased the background and resulted in the detection of gas tag activation products for the first time. Improvements in the computer data handling programs have been made which enhance the system's capability for "fingerprinting" failure types and failure locations.
- The fabrication and installation of the lab sparger was completed. Following completion of lab wiring changes the system will be ready for use in support of the EBR-II sparger development program for use in Run-Beyond-Clad-Breach-Testing. The reactor sparger design for EBR-II was completed and the sparger program reviewed in detail with ANL.
- Initial evaluation of the capabilities of the FFTF-FFM system to meet CRBR requirements was completed. It is clear that the present detection and location systems will not perform adequately in the presence of all types of on-line gas leakers (estimated maximum 60 pins/year) unless one can assume all leakers release gas continuously

after the first release. Unfortunately, the data required to substantiate such an assumption will not be available before early 1977.

Cover Gas Monitoring System - D. W. Archer, F. E. Holt, W. C. Miller

Objectives

This task provides for the development, design, procurement and implementation of the FFTF FFM Cover Gas Monitoring (CGM) System.

Summary of Accomplishments

- All equipment procurements are presently on schedule.
- A major portion of all vendor-generated fabrication documentation has been submitted for HEDL review.
- Equipment additions and modifications to the FFTF CGM Failure Characterization Subsystem Spectrometer/Computer have been initiated.
- The design of the System 94 Liquid Nitrogen Supply System was completed and all design documentation submitted for HEDL Engineering approval.

Discussion

Almost the total emphasis of this task in the last quarter has been on equipment procurement. All three major equipment suppliers completed a majority of the documentation required before fabrication could begin. Procedures for fabrication, testing, and quality assurance were submitted and approved. Drawings such as panel layouts and electrical schematics were similarly submitted and approved. With fabrication initiated late in this quarter, delivery of all equipment except the spectrometer/computer is forecasted to occur before the end of this calendar year as scheduled.

The majority of equipment items required for the upgrading of the existing spectrometer/computer were received and installed in the system. Included are a memory expansion to 32K words, the addition of a high speed

paper tape reader/punch system, additional interfaces, and switching and multiplexers to allow the equipment to process data from as many as six gamma detectors. Software development on this equipment will continue until delivery to the FFTF Site in 1977.

The System 94 Liquid Nitrogen Supply System is an automatic supply system to maintain liquid nitrogen cooling of the CGM gamma detectors and the gas tag sampling charcoal trap. Delivery of this equipment should occur early in 1976.

EBR-II Prooftesting - D. V. Archer, F. E. Holt, and W. C. Miller

Objectives

This task provides design verification and performance evaluation of the Fuel Failure Monitoring System components on an operating fast reactor.

Summary of Accomplishments

- The semi-annual calibration check of the Ge(Li) prooftest's counting equipment was completed.
- During two fission gas escapes the Ge(Li) prooftest experiment clearly measured gas tag activation products.
- The computer program which decodes the spectrometer data from the Ge(Li) prooftest was improved and modified to include: a zero and gain check and correction subroutine; a smoothing function in the plot subroutine; the addition of terms to correct counts from the analysis to counts/time-cm³ over the reactor sodium.

Discussion

The Ge(Li) Prooftest experiment installed at EBR-II is testing the effectiveness of the FFTF's cover gas sampling column and a portion of the electronic pulse height analysis equipment.

The prooftest was on-line and operating throughout most of this period and monitored the gas escapes that occurred in June and July. During

August the counting equipment was given its first semi-annual specification check which resulted in replacing a preamplifier. The replaced component had been exhibiting spurious noise pulses and was sensitive to microphonics. A spare preamplifier was installed.

During the fission gas escape that occurred in June, the system clearly measured the gas tag activation products Xe-125 and Xe-127. Figure 27 shows the system's response for four of the measured nuclides versus time. Figures 28 through 31 are individual spectra of the several hundred obtained during this period which show the appearance and increase of the appropriate gamma lines.

These measurements confirmed the validity of the FFTF absorber pin failure design by showing that the column-detector combination could indeed measure escaped tag gas. Since absorber pins (B_4C) do not contain fission products but do contain tag gas for location purposes, an increase in the activity of 188 KeV from Xe-125 without an increase in fission product activity will signal an absorber pin failure.

The data reduction programs which decode the Ge(Li) proof test data were modified and improved during this period. Gamma energy spectra are stored on magnetic tape which is sent weekly to HEDL for analysis. The program was modified by redefining most of the background correction regions from one channel band to two in order to compensate for sloped baselines. Improvements that were made to the program include: the addition of subroutines that checked the linearity of the system and made corrections as needed; the addition of a simple data smoothing subroutine which eliminated some of the sharp discontinuities that occur as a result of statistical variations; and the capability of correcting counting data to specific activities.

The linearity of the pulse height advantage data is checked by comparing the peak channels for Xe-133 (81 KeV) and Xe-135 (608 KeV) against their expected channel numbers, 243 and 1824 respectively. Using the formula for a straight line, corrected channel numbers are calculated as required and used in subsequent arithmetic manipulations.

INITIAL SEQ # 57 DATE:

6/13/75 TIME 0:5

COUNTING TIME 700 SECONDS

XE-133 XE-135

XE-125

XE-135

MAX. DATA VALUE = 553002.0

FIGURE 27. Ge(Li) Proof test Responses to Gas Escapes.

HEDL 7602-57.1

COUNTING TIME 700 SECONDS
MAX. DATA VALUE = 5149.0

SEQUENCE # 100 DATE: 6/13/75 TIME 9:56

FIGURE 28. Gamma Energy Spectrum of EBR-II Cover Gas Prior to Gas Escape.

FIGURE 29. Gamma Energy Spectrum of EBR-II Cover Gas Onset of First Gas Escape.

FIGURE 30. Gamma Energy Spectrum of EBR-II Cover Gas Prior to Second Gas Escape.

FIGURE 31. Gamma Energy Spectrum of EBR-II Cover Gas Following Second Gas Escape.

The statistical variations around low countrates give rise to wide swings in the values obtained. For plotting purposes each point is compared with the previous analysis' point and its standard deviation. If the new data point is within the expected deviation, an average of the two is plotted. If the new data point is outside the standard deviation, the new point is plotted.

The data obtained by the system are expressed as counts per time since the detector monitors gamma rays emitted by the sampling column. Those countrates are converted to specific activities by including the appropriate parameters into an equation which yields a table of conversion factors versus nuclide. The parameters are: gas flow rate onto the column; gas absorption factors; EBR-II cover gas transit time to the proof test; and cover gas volume. Each nuclide can now be expressed in units of counts/time-cm³ over the reactor sodium.

Sodium Sparger Development - R. Kolowith, F. E. Holt, and

G. W. Mettler

Objective

This task provides for the development, design, procurement and implementation of the Sodium Sparger System for reactor application. The work currently includes the design and testing of a Lab Sparger and the design of a batch type sparger for EBR-II application.

Summary of Accomplishments

- Installation of the Lab Sparger System has been completed.
- Sparger efficiency tests are scheduled to begin in October.
- A design review of the EBR-II Sparger was completed.

Discussion

A prototype batch sparger has been installed in the EDL-146 Mezzanine Area of the 324 Building. The main system consists of a static-pot sodium sparge chamber, an argon gas system for degassing the sodium and a Ge(Li) counting system for determining sparge efficiency. Auxiliary systems include leak detection, fire suppression, and filtered ventilation systems.

This laboratory prototype batch sparger system will be used to determine sparge efficiency in degassing ^{131m}Xe from liquid sodium. The ^{131m}Xe will be present in the sodium as single atoms born through decay of ^{131}I . Testing is scheduled to begin in October.

A design and program review of the EBR-II Sparger effort was held on September 18, 1975 at Idaho Falls. All electrical and mechanical drawings and the sodium piping stress analysis were presented.

Expected Accomplishments Next Quarter

- Sparge efficiency tests on ^{131}I will be completed.

CRBR Fuel Failure Monitoring System Development - Cover Gas Monitoring -

F. E. Holt

Objective

Develop and design a cover gas monitoring system that will be able to detect and characterize failed pins in the presence of earlier failures that remain in core.

Summary of Accomplishments

- The number of failed pins, N, that can remain in core and still be able to detect the $N + 1^{\text{th}}$ failure will be dependent on the dynamics of gas escapes.
- The Failure Location Subsystem's capability may be doubled by using tag activation product analysis.

Discussion

Estimated specific activities at the Cover Gas Monitoring System (CGMS) were determined for background and escaped gas cases based on the FFTF sampling column and detector design. Figure 32 is a plot of the number of failures versus large variations (10σ & 1%) around the respective countrates of escaping Xe-133, assuming steady-state equilibrium conditions. Also shown is the percent Xe-133 escaping versus its estimated countrate.

Figure 32 represents the "best" case for failure detection, that is, with the gas escape rate from N failures and the cover gas purge rate and temperature all constant. For that case and the data on Figure 32, one can see that with 20 failed pins in core that 21st pin's gas escapes would be indicated by an increase of 480 counts/sec over an existing countrate of 1.2×10^6 counts/sec. The 21st pin would have to allow about 0.7% of its stored gas to escape to attain the 480 counts/sec; but gas escape experience at EBR-II has shown that gases are evolved both continuously and intermittently. Therefore, Figure 32 cannot be used to establish the exact number of N failures that can be accommodated.

There is another parameter that will be of some value for detecting a failure, namely the measurement of escaped activation products of the tag gas that is added to all pins. The Ge(Li) Proof test data reported previously in this report showed that the nuclides Xe-125 (from Xe-124) and Xe-127 (from Xe-126) were measured during the last two gas escapes. Hence, absolute increases in the -125 and -127 as well as those relative to fission product -133 may be used to indicate the presence of an additional failure.

The -125 and -127 measurements will also benefit the Failure Location and Failure Characterization Subsystems by yielding linear power and burnup values. Ratios of -125 to -127 will determine the age of the gas while the ratio of Xe-125 to Xe-133 or Xe-134 will indicate the linear power of the failed pin. The 125/127 number decreases by a factor of ~ 30 over a period of 200 days. The ratio of -125/-133 or of -125/-134 will be strongly dependent on linear power because the release/birth ratio of the fission product gases from the fuel to the plenum is dependent on linear power.

Expected Accomplishments Next Quarter

- Determine the feasibility of using -125 and -127 measurements as "new" failure indicators.
- Determine the feasibility of using -125 and -127 measurements to increase the capability of the Failure Location Subsystem.
- Designate modifications to the FFTF CGMS to meet CRBR requirements.

HEDL 7602-57.16

FIGURE 32. Counts/Second Variations of Xe-133 Around Steady-State Counting Rates Vs. Number of Failed Pins & % Xe-133 Escaped from 1 Pin Vs. Countrate.

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