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FISSION PRODUCTS FROM FUEL DEFECT TEST AT SAXTON

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ABSTRACT

An intentional fuel cladding defect was placed in the Saxton Nuclear plant and the reactor operated at and near full power during a calendar period of 57 days. Primary coolant samples were analyzed for fission products and escape rate coefficients for the iodines and fission gases were estimated therefrom. The calculated ν 's for iodines are $2 \times 10^{-9} \text{ sec}^{-1}$, $8 \times 10^{-9} \text{ sec}^{-1}$, and $14 \times 10^{-9} \text{ sec}^{-1}$, for I^{131} , I^{133} , and I^{135} respectively. Fission gas Xe^{133} and Kr^{88} indicated average ν 's of about 25×10^{-9} .

SUMMARY

1. An experimental 3x3 fuel rod assembly, with one rod intentionally defected by drilling a 15 mil diameter hole through the cladding, was operated in the Saxton nuclear reactor for 57 days at between 10 and 23.5 MWt, accumulating 1230 MWD irradiation.
2. Coolant samples collected from the primary system were analyzed for fission product iodines and gases with the objective of defining the magnitude of escape rate coefficients.
3. Escape rate coefficients calculated from the coolant activities were as follows:

I-131	$2 \times 10^{-9} \text{ sec}^{-1}$
I-133	$8 \times 10^{-9} \text{ sec}^{-1}$
I-135	$14 \times 10^{-9} \text{ sec}^{-1}$
Xe-133	$20 \times 10^{-9} \text{ sec}^{-1}$
Kr-88	$30 \times 10^{-9} \text{ sec}^{-1}$

The values of these coefficients is less by a factor of up to 6 for the iodines and about half that for the gases used in some design calculations heretofore.

4. Computations of nuclide atom ratios in the coolant from data collected during this test are, at best, a qualitative indication that the source of fission products was a clad defect.

5. Operation of the mixed-bed, K/Borate, purification demineralizer during the test showed that fission product iodines and tramp sodium were removed with high efficiency. Fission product cesium, as expected, was not removed by the potassium form exchanger to any significant extent.

I. INTRODUCTION

Within the overall scope of the Large Closed Cycle Reactor Research and Development program at WAPD for the design and evaluation of new concepts in reactor technology, it was desirous to operate a reactor with a known clad defect in one or more fuel-tubes in a chemical shim environment. This would present an opportunity to monitor the metallurgical aspects of a defect in such an operating reactor, as well as, to measure the quantity of fission products that would be emitted to the reactor coolant. It is the latter of these programs to which this report is devoted. Other aspects of the "defect test", are reported elsewhere.^{/1}

The test of fission products release from the defect element, in general, consisted of drilling through the clad of a fuel tube, giving a 0.015 inch diameter hole, which may be conservatively compared with a weld defect. In order to obtain the highest release rate of fission products from the fuel tube, the defect was located near the middle of the rod and the rod was placed in the core center of the Saxton reactor. Thus the defect was located at the point of highest neutron flux in the reactor core. With the defect located therein, the reactor was operated for 57 days accumulating a total of approximately 1230 MWD at between 10 and 23.5 MWt. The defected rod operated at a peak power of 13.5 kw/ft and accumulated ≈ 3000 MWD/MTU peak with a peak to average ratio of 1.387. During this time, coolant samples were collected and analyzed for fission products, such as I^{131} , I^{133} , I^{135} , Xe^{133} , etc., at near steady-state operating conditions, from which calculations of escape rates from the defect can be ascertained.

In the following Sections, detailed descriptions of the test procedure are given followed by a presentation of coolant analyses and interpretation of the test results.

II. Experimental Procedures

A. The Fuel Rod and Defect

A new fuel rod, having standard dimensions for the Saxton reactor, was fabricated for the defect test. Characteristics of this rod were as given in Table 1. The fuel pellets were clad with Zircaloy-4, furnace-preoxidized on the exterior surfaces prior to drilling the defect. The defect was machine-drilled to a diameter of 0.015 inches and was located axially at the point of highest power and neutron flux. A photograph of the defected rod is presented in Figure 1. That the defect was located at the point of highest flux is illustrated in Figure 2, which shows a typical gamma-scan profile for a Saxton rod removed after operation in the same core-center positions as the defected rod.

To better illustrate the clad defect, Figure 3 shows a cross section of the rod obtained after the test during post-irradiation examination of the fuel rod. The defect is clearly seen to penetrate the clad, thus exposing the fuel to the exterior environment.

B. Placement of Defect Rod in Reactor

Following fabrication and drilling the defect, Rod 321 was positioned in a 3x3 subassembly, number 503-4-24. This subassembly contained fuel rods enclosed in a square array within a perforated can as shown disassembled in Figure 4. The subassembly contained four removable and five stationary rods, with design features as summarized in Table 2, and arranged in the 3x3 as shown in Figure 5.

TABLE 1

Dimensions and Characteristics of the Saxton Defect Fuel Tube

Code Number	321
Materials of Construction	
Clad	Zirc-4
End Plugs	Zirc-4
Fuel	UO ₂
U ²³⁵ Enrichment, percent	6.10
Physical Dimensions	
Fuel Tube	
Length (including end-plugs), inches	41.44
Outside diameter, inches, nominal	0.391
Clad thickness, inches, nominal	0.024
Inside diameter, inches, nominal	0.344
Fuel Pellets	
Outside diameter, inches, nominal	0.337
Length, inches, nominal	0.732
Total number of pellets	48
Total fuel length, inches	34.9
Pellet to clad diametrical gap, inches, nominal	0.0065
Total Weight UO ₂ , grams/% Theoretical Density	528/95
Total Weight U ²³⁵ , grams	28
Peak Power (in Saxton core center, Q = 23.5 MWt), kw/ft	13.5
Peak to Average Power in Rod,	1.387
Fuel Burnup (at end of test), MWD/MTU, in defect rod	≈ 3000 peak
Defect Diameter in Clad, inches	0.015
Defect Location (above tip of bottom end-plug), inches	17.25

4.4 X

Photograph Of 15 Mil Diameter Defect
In Zircaloy Clad Rod Following 56
E. F. P. Days Exposure In A Borated
PWR Environment

Figure 1

Figure 2

Gamma Scan - Fuel Rod #14

Figure 2

Gamma Scan of Fuel Rod
from Saxton Core Center Position
After Irradiation

COMPOSITE OF DEFECT AREA

CLAD IN AREA OF DEFECT
(ETCHED)

INTENTIONAL DEFECT IN ZIRCALOY CLAD ROD NO.321

Figure 3

DISASSEMBLED FUEL BEARING PORTION OF SAXTON SPECIAL REMOVABLE
3 x 3 FUEL ASSEMBLY.

Figure 4

TABLE 2

Summary of the Saxton Special Removable In-Pile 3 x 3 Defect Test Assembly

Saxton Core Position	Fuel Rod Number	Cladding Material	Fuel Rod Type	Number of Fuel Pellets	Enrichment (% U ₂₃₅)	UO ₂ (g)	U ₂₃₅ (g)	Defected	Previous Exposure
N-1	101	Zircaloy-4	Removable	48	6.10	528	26	No	Yes (2)
N-1	102	Zircaloy-4	Removable	48	6.10	524	26	No	Yes (2)
N-1	301	Zircaloy-4	Non-Removable	Powder	7.20	474	31	No	No
N-1	302	Zircaloy-4	Non-Removable	Powder	7.20	477	31	No	No
N-1	303	Zircaloy-4	Non-Removable	Powder	7.20	472	30	No	No
N-1	311	304 S/S	Non-Removable	Powder	7.20	540	34	No	No
N-1	312	304 S/S	Non-Removable	Powder	7.20	543	34	No	No
N-1	321	Zircaloy-4	Removable	48	6.10	528	28	Yes (1)	No
N-1	330	304 S/S	Removable	48	5.71	578	29	No	No

(1) 15 mil diameter hole at region of maximum flux.

(2) Approximately 9000 MWD/MTU.

Figure 5 Orientation of the Saxton Special Removable In-Pile 3x3 Defect Test Assembly

Legend: ZR - Zircaloy 4 Clad
 SS - 304 Stainless Steel
 N.R. - Non-Removable
 R. - Removable
 Vib. Comp. - Vibratory Compacted

The 3x3 subassembly containing the defected rod was placed in the center of a 9x9 fuel assembly, No. 503-10-1, as shown in Figure 6, which was located in the Saxton core center, position M-1, illustrated in Figure 7.

KEY

ITEMS

- A RCC TEST ELEMENT WITH PERFORATED GUIDE TUBE
- B RCC TEST ELEMENT WITH SOLID GUIDE TUBE
- C STAINLESS STEEL CLAD FUEL ROD ⁽¹⁾ WITH UO₂ PELLETS ⁽²⁾ ENRICHED TO PROVIDE 16 KW/FT PEAK POWER OUTPUT
- D STAINLESS STEEL CLAD FUEL ROD ⁽¹⁾ WITH UO₂ PELLETS ⁽²⁾ ENRICHED TO PROVIDE 13.5 KW/FT PEAK POWER OUTPUT
- E STAINLESS STEEL CLAD FUEL ROD ⁽¹⁾ WITH UO₂ PELLETS ⁽²⁾ ENRICHED TO PROVIDE 13.5 KW/FT PEAK POWER OUTPUT ⁽⁵⁾
- F ZIRCALOY CLAD FUEL ROD ⁽³⁾ WITH UO₂ PELLETS ⁽⁴⁾ ENRICHED TO PROVIDE 13.5 KW/FT PEAK POWER OUTPUT
- G ZIRCALOY CLAD FUEL ROD ⁽³⁾ WITH UO₂ PELLETS ⁽⁴⁾ ENRICHED TO PROVIDE 13.5 KW/FT PEAK POWER OUTPUT
- H ZIRCALOY CLAD FUEL ROD ⁽³⁾ WITH UO₂ PELLETS ⁽⁴⁾ ENRICHED TO PROVIDE 14 KW/FT PEAK POWER OUTPUT
- I ZIRCALOY CLAD FUEL ROD ⁽³⁾ WITH UO₂ PELLETS ⁽⁴⁾ ENRICHED TO PROVIDE 16 KW/FT PEAK POWER OUTPUT
- J BLANK SPACE PROVIDED FOR IN-CORE INSTRUMENTATION

NOTES:

- (1) ALL STAINLESS CLADDING IS .361 IN I.D. WITH .015 IN. WALL
- (2) UO₂ PELLETS FOR STAINLESS CLAD RODS ARE .357 IN. DIA.
- (3) ALL ZIRCALOY CLADDING IS .3435 IN. I.D. WITH .0237 IN. WALL
- (4) UO₂ PELLETS FOR ZIRCALOY CLAD RODS ARE .337 IN. DIA.
- (5) UO₂ FUEL PELLETS CONTAINING 100 PPM BORON

E. D. SK. 302100-B

Figure 6

End View and Arrangement of 9x9 Fuel Assembly No. 503-10-1

PLAN VIEW OF THE SAXTON CORE ILLUSTRATING
THE SAXTON SPECIAL ASSEMBLIES

Figure 7

III. REACTOR OPERATIONS AND WATER CHEMISTRY

The defected fuel rod in its subassembly was placed in the reactor core during the cold shutdown of July, 1964. The reactor coolant was heated to $\approx 530^{\circ}\text{F}$ on July 30, and power operation was begun August 12. Numerous plant difficulties caused the reactor to be shutdown several times at first, then on August 15, operation at 20 MWt was sustained for about seven days. This was followed by a shutdown, and then, beginning August 22, the reactor was operated at power without further difficulty, until the final shutdown for the defect test on October 13, 1964. Plant cooldown was accomplished on October 14, for removal of the 3x3 defect assembly.

Primary system operations and coolant chemistry are presented in Figure 8 for the period when the defect was in the reactor, i.e., July 30 to October 14, 1964. Coolant chemistry was typically that of a chemical shim plant, with ≈ 600 ppm boron as boric acid partially neutralized with $\approx 1 \times 10^{-4}$ molal alkali as potassium and lithium hydroxides. Except as noted later, there was no ion exchange purification of the primary coolant. Feed and bleed from the reactor loop through the low pressure surge tank and back to the reactor was not continuous but was initiated during each eight hour period for 10 to 30 minutes at 12 gpm. In addition to this, feed and bleed was initiated, following each water or boric acid solution addition, and continued until mixing was complete.

In addition to the chemical quantities given in Figure 8, the primary coolant contained 25-35 cc H_2 (STP)/kg coolant, and daily analyses showed there was no oxygen in the coolant, i.e., less than 0.010 ppm.

SAXTON REACTOR OPERATIONS AND COOLANT CHEMISTRY

Figure 8

SAXTON CHEMISTRY 1964

SAXTON REACTOR OPERATIONS AND COOLANT CHEMISTRY

Figure 8A

Near the end of September, the coolant demineralizer was placed in continuous service at 11.5 gpm flow for a period of about 12 hours. This was done in order to define, if possible, the contamination factors or removal efficiencies for fission products across the ion exchanger. The ion exchanger material was 6 cu. ft. of Illco NR-9 mixed bed resin regenerated in the potassium/borate forms, and was contained in a vessel having an internal diameter of 24 inches.

IV. SAMPLE ANALYSIS AND TEST RESULTS

During the period August 12 to October 25, samples of the primary coolant were collected and analyzed for fission products. Liquid coolant samples are collected at Saxton from a special sample line that connects into the loop piping between the primary coolant pump and the reactor vessel. This sample travels hot to the sample room where it may be cooled and collected in an appropriate sample vessel. For samples to be analyzed for ionic fission products such as iodine and cesium, the sample is cooled and collected in a clean polyethylene bottle. For analysis of dissolved gaseous fission products, such as xenon, the sample is isolated in a stainless steel vessel before cooling, and the vessel removed to the laboratory for gas stripping. The specific techniques used for radiochemical analysis of samples is given in Appendix A, Radiochemical Analysis of Solutions, and Appendix B, Separation and Analysis of Fission Gases from Solution Samples.

Results of coolant sampling during the test are presented in Table 3 for the period with no purification, and Table 4 for the 13 hours of coolant purification. The data of Table 3 is also presented in Figure 9 for fission products I^{131} , I^{133} , 15 m, β - γ coolant activity, reactor power, and coolant makeup. These data will be utilized to assess the escape rates of fission products from the defected fuel rod, given in following sections.

TABLE 3

Fission Product Iodine Activity in Saxton Coolant
No Purification

Date/Time 1964	Reactor Power MWt	I^{131} dpm/cc Coolant	I^{131} ν , sec ⁻¹	I^{133} dpm/cc Coolant	I^{133} ν , sec ⁻¹	I^{135} dpm/cc Coolant	I^{135} ν , sec ⁻¹	Atom Ratio I^{131}/I^{133}	Atom Ratio I^{131}/I^{135}	Atom Ratio I^{133}/I^{135}
8-15/1100	20	1.38×10^5	0.90×10^{-9}	25.3×10^5	69.3×10^{-9}			0.51		
1135	20	0.38×10^5	0.24×10^{-9}							
2300	20	0.60×10^5	0.40×10^{-9}	5.07×10^5	13.7×10^{-9}			1.10		
8-16/1100	20	1.13×10^5	0.74×10^{-9}	8.52×10^5	23.0×10^{-9}			1.23		
2300	20	1.08×10^5	0.70×10^{-9}	6.52×10^5	17.6×10^{-9}			1.54		
8-17/1100	20	1.17×10^5	0.76×10^{-9}	5.00×10^5	13.5×10^{-9}			2.07		
2300	20	1.25×10^5	0.81×10^{-9}	4.42×10^5	11.9×10^{-9}			2.62		
8-18/0752	20	1.39×10^5	0.91×10^{-9}	4.10×10^5	11.0×10^{-9}	1.33×10^5	12.6×10^{-9}	3.14	30.1	9.7
1000	20	1.35×10^5	0.88×10^{-9}	3.95×10^5	10.6×10^{-9}	0.95×10^5	9.01×10^{-9}	3.17	40.8	12.9
8-22/1800	20	13.4×10^5	8.90×10^{-9}							
8-23/1800	20	12.9×10^5	8.50×10^{-9}							
8-24/1800	20	12.4×10^5	8.19×10^{-9}							
8-25/1930	20	10.7×10^5	7.04×10^{-9}							
8-27/1100	20	6.39×10^5	4.15×10^{-9}	1.02×10^5	2.74×10^{-9}	2.28×10^5	21.7×10^{-9}	58	81	1.4
8-27/1552	20	6.02×10^5	3.92×10^{-9}	1.18×10^5	3.18×10^{-9}	1.78×10^5	16.9×10^{-9}	47	98	2.1
8-28/0825	20	6.56×10^5	4.27×10^{-9}	1.29×10^5	3.48×10^{-9}	1.00×10^5	9.49×10^{-9}	47	189	4.0
9-8 /1015	23.5	3.90×10^5	2.21×10^{-9}							
9-11/1045	23.5	4.31×10^5	2.39×10^{-9}							
9-16/1030	23.5	3.98×10^5	2.21×10^{-9}							
9-18/1030	23.5	3.94×10^5								
9-21/1000	23.5	2.71×10^5	1.77×10^{-9}							
9-23/1015	23.5	3.79×10^5	2.10×10^{-9}	0.46×10^5	1.09×10^{-9}			76		
1515	23.5	4.15×10^5	2.30×10^{-9}	0.67×10^5	1.54×10^{-9}			58		
9-24/0845	23.5	3.85×10^5	2.20×10^{-9}	0.97×10^5	2.22×10^{-9}			37		

TABLE 4

Fission Product Activity in Saxton Coolant
With Purification

Date/Time 1964*	Coolant Activity-dpm/cc x 10 ^{-3**}						System Volumes Through Demineralizer	Purification Decontamination Factors***					
	I ¹³¹	I ¹³³	I ¹³⁵	Na ²⁴	Cs ¹³⁴	Cs ¹³⁷		I ¹³¹	I ¹³³	I ¹³⁵	Na ²⁴	Cs ¹³⁴	Cs ¹³⁷
9-24/0845	385	96.6		69.5	44.2	228	0						
/1010	Purification flow begun at 11.5 gpm - Reactor at 23.5 MWt												
/1330	152	35.6		26.7	18.5	91.6	0.91	312	> 360		> 270	13	8.9
/1330AX	0.487	N.D.		N.D.	1.4	10.3							
/1645	70	23.1		15.9	8.31	40.3	1.8	180	> 280		> 160	5	5.1
/1645AX	0.390	N.D.		N.D.	1.65	7.9							
/1930	31.2	14.9		10.9	4.12	25.8	2.6	428	> 150		9	2.6	3.1
/1930AX	0.073	N.D.		1.18	1.61	8.24							
/2200	19.1	12.3	11.3	1.01	2.87	14.0	3.3	277	> 120	> 110	> 10	2.4	1.8
/2200AX	0.069	N.D.	N.D.	N.D.	1.21	7.61							
/2210	Purification flow ended												
9-25/0835	38.3	50.5	19.1	33.5	2.88	13.7	-						

* AX - after ion exchanger sample - other samples were collected from the primary coolant.

** N.D. - not detected in sample within precision of measurement - assumed for DF calculations to be less than 100 dpm/cc.

*** Decontamination factor - inlet activity/outlet activity.

Figure 9

V.

MATHEMATICAL ANALYSIS OF FISSION PRODUCT DATA

A simple model for escape of fission products from a defective fuel element is presented, which permits analysis of the data given in Table 8. The model assumes ideal conditions in most every parameter chosen, and should, therefore, not be construed as a display of rigorous equations which take into account all variables. For example, use of the constant ν , the escape rate coefficient, is made without regard to a possible variation in its value with reactor power or temperature. This coefficient is simply an "apparent" release-rate, whether the release is controlled by defect size or by diffusion of fission products from the fuel pellets, or both. (Amplification of the definition of the escape rate coefficient can be found in the literature^(2,3)) A glossary of terms used in the model follows:

- N = total atoms of nuclide
- F = fission rate in defective element, fissions/sec
- P_D = defect power, per cent
- P_t = reactor power, watts thermal
- Y = fission yield of primary fission product
- λ = decay constant, sec^{-1}
- ν = escape rate coefficient, sec^{-1}
- k_p = purification constant for fission product, sec^{-1}
- k_l = leakage constant of gas from coolant, sec^{-1}
- t = time, seconds
- F_{IX} = ion exchange flow rate, gpm
- V_T = total coolant volume (reduced to STP), gallons
- F_P = spray flow to pressurizer, gpm
- A = coolant specific activity for nuclide, dpm/ml

Subscripts

A = primary fission product

B = daughter product

Superscripts

f = in fuel or fuel tube

c = in coolant

From Table 5 it is seen that the decay chains may be conveniently divided in terms of "primary fission products" and "daughter products". The latter arise from decay of primary fission products, and are assumed to be produced in no significant yield by the fissioning process. Thus, the following nuclides have been so divided:

<u>Primary Fission Products - Yield, %</u>		<u>Daughter Products</u>
20.8 h I ¹³³	6.9	5.27 d Xe ¹³³
8.05 d I ¹³¹	3.1	none
6.7 h I ¹³⁵	6.1	9.2 h Xe ¹³⁵
2.8 h Kr ⁸⁸	3.57?	17.8 m Rb ⁸⁸

As seen from Table 5, there is some uncertainty in the yield of Kr⁸⁸ since it appears in the 89 decay scheme, as well as, the 88 scheme. Moreover, the direct yield of Kr⁸⁸ is assumed to be equal to the total chain yield which is reported only for Sr⁸⁸.

TABLE 5

FISSION PRODUCT DECAY SCHEMES *

a. 133 b. 131 c. 135 d. 88 e. 89

*

S. Katcoff, "Fission Product Yields from Neutron-Induced Fission",
NUCLEONICS, 18, 201 (1960).

NOTE: Experimental fission yields, in per cent. are underlined.

With the foregoing as a basis, therefore, a model for the escape of fission and daughter products from the defective fuel element can be constructed. The mechanism is visualized as follows:

where primary fission products (A) are produced from fission at a yield, Y, and undergo decay, λ , and escape to the coolant, v . The daughter product (B) follows a similar model. Both the primary fission product and the daughter product can be removed from the coolant by either purification, or by leakage. The expressions which describe the model of fission product release are derived in the following:

IN FUEL TUBE

PRIMARY FISSION PRODUCT (A)	DAUGHTER PRODUCT (B)
$\frac{d N_A^f}{dt} = F Y_A - N_A^f \lambda_A - N_A^f v_A = 0,$ <p>at steady state.</p> $N_A^f = \frac{F Y_A}{(\lambda_A + v_A)} \quad (1)$	$\frac{d N_B^f}{dt} = N_A^f \lambda_A - N_B^f \lambda_B - N_B^f v_B = 0, \text{ at steady state.}$ <p>Combine with (1)</p> $N_B^f = \frac{F Y_A \lambda_A}{(\lambda_A + v_A)(\lambda_B + v_B)} \quad (2)$

IN COOLANT

$$\frac{d N_A^c}{dt} = N_A^f v_A - N_A^c \lambda_A - N_A^c k_p = 0,$$

at steady state.

$$N_A^c = \frac{N_A^f v_A}{(\lambda_A + k_p)} \quad (3)$$

$$\frac{d N_B^c}{dt} = N_B^f v_B + N_A^c \lambda_A - N_B^c \lambda_B - N_B^c k_1 = 0,$$

at steady state.

$$N_A^c = \frac{N_B^c (\lambda_B + k_1)}{\lambda_A} - \frac{N_B^f v_B}{\lambda_A} \quad (4)$$

For the quantity of primary and daughter products in the coolant, first combine (1) and (3) to obtain

$$N_A^c = \frac{F Y_A v_A}{(\lambda_A + v_A)(\lambda_A + k_p)} \quad (5)$$

and (2) and (4) to obtain

$$N_A^c = \frac{N_B^c (\lambda_B + k_1)}{\lambda_A} - \frac{F Y_A v_B}{(\lambda_A + v_A)(\lambda_B + v_B)} \quad (6)$$

then combine (5) and (6), which yields:

$$N_D^c = \frac{F Y_A \lambda_A}{(\lambda_B + k_1)(\lambda_A + v_A)} \left[\frac{v_A}{(\lambda_A + k_p)} + \frac{v_B}{(\lambda_B + v_B)} \right] \quad (7)$$

Thus, equations (5) and (7) give the total primary or daughter nuclide in the coolant in terms of constants which either may be assumed, or may be determined by independent measurement. The equations are very useful in estimating the fissions affected by the defect, and the ratios of escape rate coefficients for various nuclides, provided radiochemistry data is obtained during periods of steady-state operation. For example,

equation (5) yields the fissions affected, after rearrangement to,

$$F = \frac{N_A^C (\lambda_A + \nu_A)(\lambda_A + k_p)}{\nu_A \nu_A} \quad (8)$$

provided, of course, N_A^C is determined at steady state, and an estimate is made for the escape rate coefficient, ν_A . Conversely, if the defect power or fissions-affected are known, an estimate of the escape rate coefficient may be made by rearrangement of equation (5) to

$$\nu_A = \frac{\beta \lambda_A}{1 - \beta}, \text{ where } \beta = \frac{N_A^C (\lambda_A + k_p)}{F Y_A} \quad (9)$$

Now, the ratio of any two primary fission products in the coolant, at steady-state, is given from equation (5) as

$$\frac{N_{A1}^C}{N_{A2}^C} = \frac{Y_{A1} \nu_{A1}}{Y_{A2} \nu_{A2}} \times \frac{(\lambda_{A2} + \nu_{A2})(\lambda_{A2} + k_{p2})}{(\lambda_{A1} + \nu_{A1})(\lambda_{A1} + k_{p1})} \quad (10)$$

The ratio of daughter to primary products in the coolant is obtained from equations (5) and (7), which, after collecting terms, gives

$$\frac{N_B^C}{N_A^C} = \left[\frac{\lambda_A}{(\lambda_B + k_1)} + \frac{\nu_B (\lambda_A + k_p) \lambda_A}{\nu_A (\lambda_B + k_1) (\lambda_B + \nu_B)} \right] \quad (11)$$

Moreover, it is readily seen that the escape rate coefficient of either the primary product or the daughter product may be determined, relative to the other, by rearrangement of equation (11) to the following

$$\nu_A = \frac{\nu_B}{\alpha (\lambda_B + \nu_B)} \quad (12)$$

or

$$v_B = \frac{\alpha \lambda_B}{\left(\frac{1}{v_A} - \alpha\right)} \quad (13)$$

where

$$\alpha = \frac{N_B^c (\lambda_B + k_1) - N_A^c \lambda_A}{N_A^c \lambda_A (\lambda_A + k_p)}$$

The equations given above should represent most of those which are needed to analyze a defect from the amount of activity appearing in the coolant at steady state. A summary of these equations is given in Table 6.

TABLE 6

Summary of Equations Used for Defect Analysis

E Q U A T I O N	Assumptions*
<p>1. Fissions Affected</p> $F = \frac{N_A^C (\lambda_A + \nu_A)(\lambda_A + k_P)}{Y_A \nu_A} \quad (8)$	ν_A
<p>2. Coolant Activity</p> <p>a. Primary Fission Product</p> $N_A^C = \frac{FY_A \nu_A}{(\lambda_A + \nu_A)(\lambda_A + k_P)} \quad (5)$	F, ν_A
<p>b. Daughter Product</p> $N_B^C = \frac{FY_A \lambda_A}{(\lambda_B + k_1)(\lambda_A + \nu_A)} \left[\frac{\nu_A}{(\lambda_A + k_P)} + \frac{\nu_B}{(\lambda_B + \nu_B)} \right] \quad (7)$	F, ν_A, ν_B
<p>3. Coolant Activity Ratios</p> <p>a. Primary to Primary Fission Products</p> $\frac{N_{A1}^C}{N_{A2}^C} = \frac{Y_{A1} \nu_{A1} (\lambda_{A2} + \nu_{A2})(\lambda_{A2} + k_{P2})}{Y_{A2} \nu_{A2} (\lambda_{A1} + \nu_{A1})(\lambda_{A1} + k_{P1})} \quad (10)$	ν_{A1}, ν_{A2}
<p>b. Daughter to Primary Fission Product</p> $\frac{N_B^C}{N_A^C} = \frac{\lambda_A}{(\lambda_B + k_1)} + \frac{\nu_R \lambda_A (\lambda_A + k_P)}{\nu_A (\lambda_B + k_1)(\lambda_B + \nu_B)} \quad (11)$	ν_A, ν_B
<p>4. Escape Rate Coefficients</p> <p>a. Primary Fission Product</p> $\nu_A = \frac{\beta \lambda_A}{(1 - \beta)}, \text{ where } \beta = \frac{N_A^C (\lambda_A + k_P)}{FY_A} \quad (9)$ <p>b. Ratio of Daughter to Primary Fission Product</p> $\nu_A = \frac{\nu_B}{\alpha (\lambda_B + \nu_B)} \quad (12)$ <p>or</p> $\nu_B = \frac{\alpha \lambda_B}{(\frac{1}{\nu_A} - \alpha)} \quad (13)$ <p>where</p> $\alpha = \frac{N_B^C (\lambda_B + k_1) - N_A^C \lambda_A}{N_A^C \lambda_A (\lambda_A + k_P)}$	F ν_B ν_A

* To solve a given equation, it is expected that all variables can be measured or approximated except those stated in this column, and in some cases, these too may be assumed.

VI. ANALYSIS OF DATA COLLECTED DURING DEFECT TEST

In the preceding section, equations were developed which will permit an analysis of fission product data collected during the defect test. For the most part, calculations can be made to assess the apparent escape rate coefficient, v , for fission product iodines, based on coolant activities during periods of steady-state reactor operation. These calculations follow:

The escape rate coefficients for primary fission products, such as I^{131} , I^{133} , or I^{135} , are determined with Eqn. 9, i.e.,

$$v_A = \frac{\beta \lambda_A}{(1-\beta)} \quad \text{where} \quad \beta = \frac{N_A^C (\lambda_A + k_p)}{F Y_A} \quad (9)$$

To solve this equation, use is made of the quantities given in Table 7, which, for the defect test data, give

- a. F , fissions affected (considering one defective fuel rod at 13.5 kw/ft peak, and peak to average of 1.387, re. Table 1)

$$\begin{aligned} \text{Defect Power} &= \frac{13.5 \text{ kw/ft (peak at 23.5 MWt)}}{(1.387 \text{ peak/average})} \times \frac{34.9 \text{ in. fuel}}{(12 \text{ in/ft})(23.5 \text{ MWt})} \\ &= 1.205 \text{ kw/MWt} \\ F &= \frac{(1.205 \text{ kw/MWt})(10^{-6} \text{ MWt/watt})(10^3 \text{ watts/kw}) P_t}{(3.2 \times 10^{-11} \text{ watt-sec/fission})} \\ &= 3.76 \times 10^7 P_t \end{aligned}$$

- b. N_A^C , atoms of nuclide in coolant,

$$N_A^C = \frac{(A_A^C)(V_T)}{(\lambda_A)(60)}$$

TABLE 7
Constants Used in Fission Product Analysis

Nuclide	Half-Life	Fission Yield Per Cent	Decay Constant, sec ⁻¹	Cross Section σ, barns
I-131	8.05 d	≈ 3.1	9.96×10^{-7}	-
I-133	20.8 h	6.9	9.25×10^{-6}	-
I-135	6.7 h	6.1	2.87×10^{-5}	-
Xe-133	5.27 d	0	1.52×10^{-6}	-
Xe-135	9.2 h	≈ 0	2.09×10^{-5}	2.7×10^6
Kr-88	2.8 h	3.57 ?	6.87×10^{-5}	-

Purification Constants

For Ion Exchange or Leakage $k_p = \frac{F_{IX}}{V_T \times 60}$

where F_{IX} = ion exchange or leak flow rate, gpm

V_T = total coolant volume, gallons

At Saxton,

$$k_p = \frac{F_{IX}}{2400 \text{ gal} \times 60 \text{ s/m}} = 7 \times 10^{-6} F_{IX} \text{ sec}^{-1}$$

* For Gaseous Leakage (100% from pressurizer) $k_l = \frac{F_p}{V_T \times 60}$

where F_p = spray-line flow to pressurizer, gpm

V_T = total coolant volume, gallons

At Saxton,

$$k_l = \frac{\approx 0.2}{2400 \times 60} = 1.4 \times 10^{-6} \text{ sec}^{-1}$$

* Since about August, 1963, a sizeable gaseous leak from the main coolant has been measured at Saxton, and it is assumed, herein, that gases carried to the pressurizer via the spray line will leak from the system.

where A_A^c = nuclide activity, dpm/ml coolant

V_T = total coolant volume, ml

= 9.1×10^6 ml at Saxton

$$N_A^c = \frac{A_A^c}{\lambda_A} (1.52 \times 10^5)$$

c. k_p , purification or leakage constant,

During the period of sampling, re. Figure 9, of August 15 to September 25 (41 days) the total makeup for leakage was about 1815 gallons, for an average of 44 gal/day. Thus,

$$\begin{aligned} k_p &= \frac{44 \text{ gpd} \times 3.785 \text{ ml/gal}}{9.1 \times 10^6 \text{ ml} \times 8.64 \times 10^4 \text{ s/d}} \\ &= 2.12 \times 10^{-10} \text{ sec}^{-1} \end{aligned}$$

This value, compared to the decay constants of any of the iodine isotopes of interest, is negligible and may be omitted, therefore, in the computation of β of Eqn. (9).

With the above substitutions, β is reduced to

$$\beta = \frac{A_A^c}{P_t Y_A} (4.04 \times 10^{-3})$$

Using this value for β , Eqn. (9) was solved for v_A for each value of I^{131} , I^{133} , and I^{135} and these are given in Table 3.

Some additional samples of coolant were collected to obtain the concentration of fission product gases resulting from the defect. The techniques used to collect and analyze the samples are given in Appendix B. The data obtained is given in Table 8, showing

TABLE 8

Fission Product Gases in Saxton Coolanta. Xe^{133}

Sample Date/Time	Coolant Activity, dpm/cc		ν of I^{133} sec ⁻¹	ν of Xe^{133} sec ⁻¹	Remarks
	Xe^{133}	I^{133}			
8-18-64/0755	4.89×10^4	4.10×10^5	-	-	20 MWt power
9-23-64/1715	2.34×10^6	6.68×10^4	1.54×10^{-9}	1.57×10^{-8}	23.5 MWt power
9-24-64/0900	3.14×10^6	9.66×10^4	2.22×10^{-9}	2.29×10^{-8}	23.5 MWt power
9-24-64/1719	1.95×10^6	1.23×10^4	2.22×10^{-9}	1.15×10^{-8}	{ After 1.92 system vols. through IX purification 10.4 hours after end of IX purification
9-25-64/0835	2.44×10^6	5.05×10^4	$2.22 \times 10^{-9*}$	3.43×10^{-8}	

b. Xe^{135}

Sample Date/Time	Coolant Activity, dpm/cc		ν of I^{135} sec ⁻¹	ν of Xe^{135} sec ⁻¹	Remarks
	Xe^{135}	I^{135}			
8-18-64/0755	5.23×10^4	1.33×10^5	↑	↑	(See Above)
9-23-64/1715	3.99×10^5	-	↓	↓	
9-24-64/0900	4.11×10^5	-	Not	Not	
9-24-64/1719	1.33×10^5	-	Calcd.	Calcd.	
9-25-64/0835	1.43×10^5	-	↓	↓	

c. Kr^{88}

Sample Date/Time	Kr^{88} Activity	Reactor Power MWt	ν of Kr^{88} sec ⁻¹
9-23-64/1600	5.70×10^4	23.5	1.92×10^{-8}
9-24-64/0840	12.7×10^4	23.5	4.26×10^{-8}

* Assuming 4.26×10^{-9} best value for ν of I^{133} from previous data.

activity concentrations for Xe^{133} , Xe^{135} , and Kr^{88} . For Xe^{133} , the escape rate coefficients were computed from Eqn. (13), using the observed data for parent I^{133} . No escape coefficients were computed for Xe^{135} since the data obtained for concurrent I^{135} are too meager and uncertain. The escape rate coefficient for parent Kr^{88} was computed using Eqn. (9) with constants relating to defect power used previously, above.

Radiochemical analyses of the primary coolant and purification demineralizer effluent during one day of purification are presented in Table 4. From this data, the demineralizer efficiency was computed for fission products iodine and cesium, and tramp sodium. During this period of demineralizer purification, the flow rate was maintained at 11.5 gpm. The demineralizer consisted of 6 cu. ft. of Illco NR-9, mixed-bed resin, regenerated in the potassium-borate forms. At 11.5 gpm flow rate, the cross-sectional flow is 3.7 gpm/ft² and the bed depth is nominally 2 ft.

Decontamination factors for nuclides (IX inlet activity/IX outlet activity) are given in Table 4, where data was collected. In many instances, the outlet activity of a particular nuclide was at or below the limit of detection, and for these a minimum D.F. is tabulated assuming a detection limit of 100 dpm/cc, in effluent samples.

The data for I^{131} and I^{133} activities in the coolant, which represent ion exchange inlet samples in Table 4, may be analyzed as a function of time to further support the mechanism of fission

product release from the defected fuel rod. As given earlier, the change in nuclide inventory in the coolant is expressed by,

$$\frac{d N_A^C}{dt} = N_A^f v_A - N_A^C \lambda_A - N_A^C k_p$$

which may be rearranged and integrated between limits of t_0 for N_{Ao}^C and t for N_A^C . The result is,

$$\frac{N_A^C}{N_{Ao}^C} = \frac{N_A^f v_A}{N_{Ao}^C \beta} + e^{-\beta t} \left(1 - \frac{N_A^f v_A}{N_{Ao}^C \beta} \right)$$

where $\beta = (\lambda_A + k_p)$

This equation may be simplified further by noting that at steady state, at the time of start of demineralization, $N_A^f v_A = N_{Ao}^C \lambda_A$. Thus,

$$\frac{N_A^C}{N_{Ao}^C} = \frac{\lambda_A}{\beta} + e^{-\beta t} \left(1 - \frac{\lambda_A}{\beta} \right)$$

Parameters used during the period of ion exchange purification were substituted into the equation to obtain the theoretical variation of I^{131} and I^{133} with time. This is shown in Figure 10, together with the observed data for these two nuclides. For these calculations, the demineralizer efficiency to remove the iodine nuclides was taken to be 100 percent. Also shown in Figure 10 is the time variation in concentration due to ion exchange of a hypothetical substance with no decay and without a continuous source to the coolant, included for comparison.

Figure 10

VII. DISCUSSION OF RESULTS

The mathematical model for the escape of fission products from a defected fuel rod, presented in Section V, is considered valid and applicable to the analysis of data from pressurized water reactors. The model does use an escape rate coefficient, ν , as a general parameter, and assumes this to be constant regardless of defect position in the fuel rod cladding, or regardless of the fuel temperature or location in the core. With only one known defect in the core, the conditions which might affect ν were fixed, and since these conditions were the most severe that could be obtained at Saxton, the calculated ν 's are conservatively higher than might be obtained from a core-average defect.

The value of ν for fission product iodines is best ascertained from the last two days of sampling (re. Table 3), for which period the coolant activity was nearest to steady state. The average ν 's are:

$$\text{I-131 (8.05d)} \quad \nu = 2.20 \times 10^{-9} \text{ sec}^{-1}$$

$$\text{I-133}_a \text{ (20.8h)} \quad \nu = 1.62 \times 10^{-9} \text{ sec}^{-1}$$

Owing to the short half life of I-133 and I-135, other data from Table 3 may be selected to obtain an average ν for periods after the reactor had operated steadily, i.e., where these nuclide activities would be near steady-state. Figure 9 indicates that data of 8-18, 8-27 and 8-28 would be near steady-state, for which average ν 's are:

$$\text{I-133}_b \text{ (20.8h)} \quad \nu = 6.2 \times 10^{-9} \text{ sec}^{-1}$$

$$\text{I-135 (6.7h)} \quad \nu = 13.9 \times 10^{-9} \text{ sec}^{-1}$$

These data indicate a variation in ν between the same nuclides of different half-lives. This is, a priori, surprising, considering the chemical and physical sensitivity of isotopes, however it can be shown^{2,3}

that the escape rate coefficient is related to the diffusion coefficient as follows:

$$v = 3D\sqrt{\lambda}$$

where $D(\text{sec}^{-\frac{1}{2}})$ is a modified mean diffusion coefficient for the nuclide in UO_2 . This relation results from the assumption that the escape is controlled by diffusion in the fuel pellet and shows that the resulting v will be directly proportional to $\sqrt{\lambda}$. Average data for v , given above, can be compared for consistency after being normalized by a divisor of $\sqrt{\lambda}$, as follows:

<u>Nuclide</u>	<u>Apparent v, sec^{-1}</u>	<u>λ, sec</u>	<u>$v/\sqrt{\lambda}$</u>
I-131	2.20×10^{-9}	9.96×10^{-7}	2.2×10^{-6}
I-133a	1.62×10^{-9}	9.25×10^{-6}	0.52×10^{-6}
I-133b	6.2×10^{-9}	9.25×10^{-6}	2.0×10^{-6}
I-135	13.9×10^{-9}	2.87×10^{-5}	2.6×10^{-6}

The result shows that escape rate coefficients, of the same element at least, are consistent when compared on the basis of diffusion coefficients. This agrees with the thesis of Frank, Allison, and others wherein the release of fission products to the reactor coolant via a defect is controlled in large measure by diffusion in the fuel.

A similar appraisal of the apparent escape rate coefficients for fission gases may be made, as above. Table 8 gives the calculated values of v for Xe-133 and Kr-88. The averaged values for these v 's, considering that all data represent the steady-state condition, are:

<u>Nuclide</u>	<u>Apparent v, sec^{-1}</u>	<u>λ, sec^{-1}</u>	<u>$v/\sqrt{\lambda}$</u>
Xe-133	2.11×10^{-8}	1.52×10^{-6}	17.5×10^{-6}
Kr-88	3.09×10^{-8}	6.87×10^{-5}	3.8×10^{-6}

While the normalized ν for Kr-88 is in reasonable agreement with the values computed for the iodines, the Xe-133 is significantly different. There is no apparent reason to explain the discrepancy, although one would not necessarily expect the behavior of iodines and fission gases to be the same. If the rate controlling step for the release of fission products from a defect is diffusion in the fuel pellet, as pointed out by Allison and Frank, a difference in the release of halogens versus rare gases would not be altogether unexpected. However, since the difference in normalized ν 's for Xe and Kr is significant, one may speculate that this results from the mode of calculation, wherein Kr is treated as a primary product of fission, while the ν of Xe is computed with reference to the parent I-133 coolant activity and escape rate coefficient.

It is of interest that escape rate coefficients for gaseous fission products Xe and Kr were measured to be higher by a factor of 10 than the iodine nuclides. This is somewhat different than has been reported by other investigators, such as Frank⁴ in early studies at BAPL, who reported that Cs, I, Xe, Kr, Rb, and Br each have an escape rate coefficient of $1.3 \times 10^{-8} \text{ sec}^{-1}$. Some of his data, however, would indicate the coefficient for gases to be a factor of 10 higher than the iodines. Subsequent work by Frank² with a defected UO_2 fuel rod indicates the escape rate coefficient of Xe-133 and Xe-135 is nearly the same as I-131 and I-133, while that of the kryptons is higher than that of the iodines by a factor of 5 to 35.

As pointed out above, the apparent escape rate coefficient for an element will vary with the half-life of the particular nuclide. Moreover, Allison and Rae have shown recently that fuel element length, as well as fuel

temperature, UO_2 crystallite size and density, will significantly alter the coefficient value. Needless to say, in view of the large number of variables which may affect the escape rate of fission products from a defect, the extrapolation of this and other published data to predict releases from plants yet to be operated is tenuous, at best. Nonetheless, for UO_2 fuel rods which will operate at power ratings no higher than this particular defect assembly (13.5 kw/ft), and are similar dimensionally (re. Table 1), the escape rate coefficients calculated herein are believed adequate.

One additional consideration should be made with regard to the calculated, apparent escape rate coefficients. It is entirely possible that the intentional defect was accompanied by other defects in the reactor core. The Saxton core has indicated the presence of other defects during operation prior to this defect test. Whether or not the unknown defect was with the assembly removed to make way for the defect assembly is not known. Should there have been an additional defect somewhere in the core during this test, the values for v would be conservatively high - the proper direction for design purposes.

The data presented in Table 4, on which Figure 10 was based, indicates good agreement between theory and observation for the apparent transport of fission products from a defect. There is a marked difference between system clean-up of a substance that is not continually fed to the coolant and clean-up of fission products arising from the defect.

The decontamination factors given in Table 4 for various nuclides are very much as expected. All nuclides were efficiently removed from the coolant with the exception of cesium. This element is not favorably exchanged by potassium-form ion exchange resins. In other cases, cesium is efficiently removed by hydrogen- or lithium-form resins.

VIII. CONCLUSIONS

The defect test in the Saxton Reactor has been completed with a number of important results. In retrospect, a much longer operating time with the defect in-core would have been preferred. However, the data obtained has permitted estimates of escape rate coefficients of the halogen and rare gas fission products. Heretofore, design calculations have used very conservative escape rate coefficients for these fission products ($1.3 \times 10^{-8} \text{ sec}^{-1}$ for iodines and $6.5 \times 10^{-8} \text{ sec}^{-1}$ for gases) but the results of this test indicate the magnitude to be a factor of up to 10 for iodine. A summary of the escape rate coefficients averaged from data collected during this test is as follows:

<u>Nuclide</u>	<u>$\nu, \text{ sec}^{-1}$</u>
I-131	2×10^{-9}
I-133	8×10^{-9}
I-135	14×10^{-9}
Xe-133	20×10^{-9}
Kr-88	30×10^{-9}

These values may be used for design purposes so long as it is assumed that a reactor core will have operating characteristics (fuel power rating) and physical dimensions (fuel composition, fuel rod length, etc.) similar to those of this defect rod.

APPENDICES

A-1 Radiochemical Analysis of Solutions

A-2 Separation and Analysis of Fission Gases from Solution

APPENDIX A-1

RADIOCHEMICAL ANALYSIS OF SOLUTION SAMPLES

Samples of the Saxton primary coolant were collected at a sample line leading directly from the primary loop. The sample stream was cooled from $\approx 530^{\circ}\text{F}$ to $\approx 80^{\circ}\text{F}$ and after sufficient purge time to cleanse the line, a batch sample was collected in a new, clean, polyethylene bottle. Aliquots of this sample were analyzed for iodine and cesium fission products after a delay of at least 10 hours. Individual analyses were performed as follows:

Iodine: to separate iodine activities from other fission and corrosion product activities in the sample, the usual, repetitive oxidation-reduction reaction with iodine carrier was performed, with a final extract of iodine into an aqueous layer from CCl_4 .^{/5} The iodine was precipitated with acidified AgNO_3 , and the AgI precipitate filtered, and mounted for analysis by gamma spectrometry.

The composite gamma-spectrum of I-131, I-133, and I-135 was analyzed by graphical-stripping methods^{/6} and confirmed by allowing the sample to decay and recounting. All activities are corrected for decay to the time of sampling.

Cesium: in, some samples, activities other than cesium were low due to long decay times, in which case, Cs^{137} and Cs^{134} were determined by direct γ -counting of a sample aliquot. The 0.662 Mev peak of Cs^{137} and the 0.601 and 0.796 Mev peaks of Cs^{134} were resolved by graphical methods combined with algebraic computation. The method of peak resolution is easily standardized through use of standard, pure Cs^{137} and Cs^{134} isotope mixtures.

For samples which contained significant quantities of other nuclides, cesium was separated from solution by the tetraphenyl boron extraction method,⁴⁷ and the resulting extract γ -counted and analyzed as above.

Sodium: although not a fission product, Na^{24} was detected in the reactor coolant (from trace Na contamination) and analyzed by direct γ -counting a sample aliquot. The quantity of Na^{24} was determined from the gamma spectrum, by integrating the area of the 2.74 Mev peak.

APPENDIX A-2

SEPARATION AND ANALYSIS OF FISSION GASES FROM SOLUTION

Primary coolant samples for gas stripping and analysis were collected in stainless steel, valve-ended vessels, of about 50 ml volume. The coolant sample stream was allowed to flow from the reactor system through the sample vessel without cooling. After purging the line and sample vessel for at least 10 minutes at ≈ 0.6 gpm, the downstream vessel valve was slowly closed, then the upstream vessel valve was closed, thus isolating the liquid/gas sample. The sample vessel was removed to the laboratory while still hot ($> 60^{\circ}\text{C}$) and connected to an evacuated, glass bulb having a volume of ≈ 60 ml. The valve between vessel and bulb was carefully opened, followed by slight opening of the other vessel valve, thus sweeping dissolved gases from the vessel to the bulb. Aliquots of the gas in the bulb were removed by hypodermic syringe and transferred to small (3 ml), evacuated, counting vials. These vials were then γ -counted using a 2 x 2 well crystal and 400 channel spectrometer. The spectrum of Xe-133 and Xe-135 was resolved by graphical stripping methods¹⁶ confirmed by decay studies.

For Kr-88, use was made of the β -activity of the Rb-88 daughter. For this, a gas aliquot was removed from the bulb, described above, and introduced into an evacuated counting vial containing 1.0 ml of 2M HNO_3 . After standing for one hour, to permit equilibrium between Kr-88 and Rb-88, the vial was vigorously shaken, opened, and an aliquot of the nitric acid was removed, dried on a planchet, and β -counted. The amount of Kr-88 in the coolant sample was then calculated from the assay of Rb-88 on the planchet.

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