**Ferret-Sand II Physics-Dosimetry
Analysis for N Reactor Pressure
Tubes 2954, 3053 and 1165 Using
a WIMS Calculated Input Spectrum**

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May 1988

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FERRET-SAND II PHYSICS-DOSIMETRY ANALYSIS
FOR N REACTOR PRESSURE TUBES 2954, 3053 AND
1165 USING A WIMS CALCULATED INPUT SPECTRUM

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SUMMARY

Neutron fluxes and fluences have been derived for N Reactor Pressure Tubes 2954, 3053 and 1165 using the FERRET-SAND II codes. The estimated uncertainty associated with the FERRET-SAND II derived values of flux and fluence ($E > 1$ MeV) for these pressure tubes is about $\pm 21\%$ (1 standard deviation).

Methods used by Battelle Columbus Laboratory (BCL) were compared to those used by Pacific Northwest Laboratory's (PNL) National Dosimetry Center (NDC) for calculating fast flux and fluence ($E > 1$ MeV). The largest difference between the BCL and PNL-NDC results is traceable to the use of significantly different values of the $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ spectral averaged cross-section for flux and fluence ($E > 1$ MeV).

The quality of neutron spectra, derived from WIMS and MCNP calculations for the pressure tube location, was evaluated by comparing measured versus calculated values of dosimetry sensor reaction rates and derived exposure parameters. The present analysis indicates that excellent agreement appears to exist between measured and calculated (MCNP based) reaction rates and exposure parameter values. However, the results of a N Reactor pressure tube physics-dosimetry experiment would be required to verify the quality of this agreement.

A preliminary analysis indicates that a reliable a posteriori pressure tube surveillance dosimetry system [using two high-energy $\{^{54}\text{Fe}(n,p)^{54}\text{Mn}$ and $^{58}\text{Ni}(n,p)^{58}\text{Co}\}$ and three low-energy $\{^{94}\text{Zr}(n,\gamma)^{95}\text{Zr}$, $^{124}\text{Sn}(n,\gamma)^{125}\text{Sn}$, and $^{59}\text{Co}(n,\gamma)^{60}\text{Co}\}$] reactions could be established and validated for adjusting past, present and future WIMS and MCNP calculations.

It is suggested that consideration be given to the analysis of pressure tube specimens for helium and boron content to see if (n,He) stable product reaction results can be used to provide an additional and independent check on the reliability of the existing and future physics-dosimetry results. Helium reaction (n,He) results provide a direct measurement of time-integrated exposure parameter values without the need for establishing a flux-level time history for a pressure tube.

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INTRODUCTION

This report is in response to a request from Westinghouse Hanford Company (WHC) that the PNL National Dosimetry Center (NDC) perform physics-dosimetry analyses ($E > 1$ MeV) for N Reactor Pressure Tubes 2954 and 3053. As a results of these analyses, and recommendations for additional studies, two physics-dosimetry re-evaluations for Pressure Tube 1165 were also accomplished. Details of these re-evaluations have been documented separately; (a)(b) however, the most important results are included here as Appendix G.

The primary objective of Pacific Northwest Laboratories' (PNL) National Dosimetry Center (NDC) physics-dosimetry work for N Reactor was to provide FERRET-SAND II physics-dosimetry results to assist in the assessment of neutron radiation-induced changes in the physical and mechanical properties of N Reactor pressure tubes.

To accomplish the primary objective, it was necessary to 1) evaluate the quality of previously calculated flux-fluence spectra for N Reactor pressure tubes, and 2) evaluate the applicability and the accuracy of the current PNL-NDC processing codes for use with long-term dosimetry results.

In addition, a preliminary assessment was made of an improved a posteriori dosimetry system for N Reactor pressures tubes. This serves as a good example of how an improved dosimetry system could be developed for other N Reactor components and for components of power-generating reactors.

The details of the methods and procedures used for the present physics-dosimetry analysis are documented in the FTR Dosimetry Handbook (Lippincott and McElroy 1983) and in the LWR Power Reactor Surveillance Physics-Dosimetry Compendium (McElroy 1985), and need not be repeated here. Briefly, however, the procedure used herein is as follows:

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- (a) McElroy, W. N., L. S. Kellogg, and A. E. Suski. FERRET-SAND II Physics-Dosimetry Analysis for the N REactor Pressure Tube 1165 Using a WIMS Calculated Input Spectrum. PNL-NDC Letter Report 3, March 28, 1988.
 - (b) McElroy, W. N., L. S. Kellogg, W. C. Morgan, and A. E. Suski. Fluence Measurement Investigations for N Reactor Pressure Tube 1165. PNL-NDC Letter Report 4, May 9, 1988.

1. End-of-irradiation specific activities were obtained on 14 to 16 dosimetry specimens taken at different axial locations along each pressure tube.
2. The TIMWH code (Lippincott and McElroy 1983, McElroy 1985) was used to derive reaction rates from the measured end-of-irradiation specific activities [for two high-energy response sensors $\{^{54}\text{Fe}(n,p)^{54}\text{Mn}$ and $^{58}\text{Ni}(n,p)^{58}\text{Co}\}$ and three low-energy response sensors $\{^{94}\text{Zr}(n,\gamma)^{95}\text{Zr}$, $^{124}\text{Sn}(n,\gamma)^{125}\text{Sn}$, and $^{59}\text{Co}(n,\gamma)^{60}\text{Co}\}$] using the N Reactor tube power-time history supplied by WHC.
3. These reaction rates and a WIMS calculated spectrum were then used as input to the FERRET-SAND II codes (Lippincott and McElroy 1983, McElroy 1985) to derive neutron flux and fluence values for the pressure tubes.

The results of the FERRET-SAND II analyses for Pressure Tubes 2954 and 3053 are presented and discussed in the Results and Discussion sections; also considered are data quality assurance issues that have had an impact on establishing the reliability of the reported physics-dosimetry results. In the Recommendations section, recommendations are presented for work that could improve the existing N Reactor physics-dosimetry program. Appendices A and B provide documentation of the work associated with the derivation of dosimetry sensor reaction rates and the measurement of dosimetry sensor specific activities and elemental concentrations for Pressures Tubes 2954 and 3053. The calculated effects of transmutations on the physics-dosimetry analysis are presented in Appendix C. Appendix D contains information provided by WHC for the analysis of Pressure Tubes 2954 and 3053. Appendix E contains information provided by WHC for the re-analysis of Pressure Tube 1165. Appendices F and G document the most important results from the initial physics-dosimetry analysis and the subsequent re-evaluation for Tube 1165.

RESULTS

REACTION RATE DATA

Appendix A lists the power-time histories (supplied by WHC) and outlines the TIMWH calculations of reaction rates for the five reactions of interest at each of 16 (Pressure Tube 2954) or 14 (Pressure Tube 3053) positions along the pressure tubes. Appendix B outlines the calculation of the specific end-of-irradiation isotope activities that were used as input to the TIMWH calculations. Matsumoto also provided (Table 3 of Appendix B, p. B.6) a comparison between the originally reported concentrations of selected elements^(a) and the concentrations measured by inductively coupled plasma spectroscopy (ICP) on specimens cut from the irradiated tubes. Except for the concentration of cobalt, there is reasonable agreement between the original results and those obtained by ICP. In particular, the ICP analyses of four specimens from Pressure Tube 2954 yields a cobalt concentration (114 ± 5 ppm) more than ten times the maximum acceptable cobalt concentration (10 ppm) given in the original purchase specification,^(b) and that listed on the original data sheet (<10 ppm).

Figures 1 and 2 show the relative reaction rates of the three low-energy neutron reactions, as a function of distance along Pressure Tubes 2954 and 3053, respectively. Note that the reaction rates are multiplied by a constant, in order to make the curves easily comparable; also note that the multiplier for the ^{59}Co reaction rate is 12 for Pressure Tube 2954 (Figure 1), but it is unity for Pressure Tube 3053 (Figure 2). The almost constant proportionality of the reaction rates, for either tube, indicates the axial uniformity of the concentrations of Co and Sn within the zirconium alloy tube. However, the fact that the ratio of the $^{59}\text{Co}/^{94}\text{Zr}$ reaction rates is a factor of about twelve lower for specimens from Pressure Tube 2954 than for Pressure Tube 3053, indicates either that ICP analysis of the cobalt concentration must be in error, or that a highly significant fraction of the current cobalt concentration was not originally present in the tube alloy. The transmutation chains which could have increased the cobalt content are

(a) Copies of the original data sheet are reproduced in Appendix D.

(b) HWS-6502 REV 2 - Specification for Zircaloy-2 Pressure Tubes for 105-N.

FIGURE 1. Pressure Tube 2954 Low-Energy Neutron Reaction Rates

FIGURE 2. Pressure Tube 3053 Low-Energy Neutron Reaction Rates

examined in Appendix C; however, the effects of these transmutation chains are much too small to account for the observed differences.

Figures 3 and 4 show the relative reaction rates of the two high-energy neutron reactions, as a function of distance along Pressure Tubes 2954 and 3053, respectively. In contrast to the almost constant proportionality of the reaction rates for the low-energy reactions, as seen in Figures 1 and 2, the ratio of the reaction rates $^{58}\text{Ni}/^{54}\text{Fe}$ decreases towards the center of the reactor.

The products of the $^{58}\text{Ni}(n,p)$ reaction (^{58}Co and ^{58m}Co) have short half-lives and large thermal-neutron cross-sections (ASTM 1987b), compared to ^{54}Mn ; thus, the end-of-irradiation specific activity of ^{58}Co in a specimen is much more dependent on irradiation details (such as thermal-to-fast flux ratios, flux intensities, and the duration and timing of shut-down periods near the end-of-irradiation) than is the specific activity of ^{54}Mn . Therefore, the reaction rates derived from ^{54}Mn specific activities can generally be considered more reliable than those derived from the ^{58}Co activities.

NEUTRON SPECTRA, SPECTRUM-AVERAGED CROSS-SECTIONS, AND FLUENCES

As a result of the uncertainty associated with the cobalt impurity content, the $^{59}\text{Co}(n,\gamma)^{60}\text{Co}$ reaction rate data were not used in the FERRET-SAND II analysis. Further, the $^{94}\text{Zr}(n,\gamma)^{95}\text{Zr}$ and $^{124}\text{Sn}(n,\gamma)^{125}\text{Sn}$ measured reaction rate data could not be directly used because the ENDF/B-V energy dependent cross-sections for these two reactions have not been processed for addition to the SAND II 620 group (SIGS) or the FERRET 53 group (LIB) dosimetry files. It is recommended that this should be accomplished for future PNL-NDC N Reactor physics-dosimetry studies.

The problems associated with obtaining reliable reaction rates for the $^{58}\text{Ni}(n,p)^{58}\text{Co}$ reaction for long-term dosimetry also precluded the use of this reaction from the FERRET-SAND II analysis. The input reaction rate for the $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ reaction was increased by a factor of 1.048, as estimated by Morgan (Appendix C), to account for the burnout of ^{54}Fe target atoms over the approximately 23 years of reactor operation. The FERRET-SAND II analysis used the 69 group WIMS-calculated flux spectrum as input. Note, however,

FIGURE 3. Pressure Tube 2954 High-Energy Neutron Reaction Rates

FIGURE 4. Pressure Tube 3053 High-Energy Neutron Reaction Rates

that the WIMS-calculated spectrum was supplied in normalized form and, thus, had to be renormalized to the measured $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ reaction rate.

The solution results (with one-sigma uncertainty estimates) for Pressure Tubes 2954 (at an axial location of 343.4 in.) and 3053 (at an axial location of 351.2 in.) are given in Table 1. These results apply equally to both pressure tubes at the designated axial distances because the measured $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ reaction rates are identical.

Table 2 shows that the ^{54}Fe reaction rate, calculated for the adjusted flux spectrum, is in agreement with the measured reaction rate; however, this result is of no particular significance, because the input spectrum was renormalized to the measured reaction rate.

The results shown in Tables 1 and 2 were then used to derive the $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ spectral-averaged cross-sections for neutrons with energies $E > 0.1$, $E > 0.18$, and $E > 1$ MeV (3.584×10^{-26} , 4.123×10^{-26} , and 9.489×10^{-26} cm², respectively). These values were, in turn, used to divide the 14 to 16 measured axial values of $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ reaction rates to derive fluence profiles for each pressure tube. The drop-off in the burn-out of ^{54}Fe target atoms (as the thermal flux level decreases) near the ends of the pressure tubes has not been factored into the present analysis. The resultant fluences, as a function of distance along the tubes, are listed in Tables 3 and 4, and are shown graphically in Figures 5 and 6.

TABLE 1. FERRET-SANO II Results for Pressure Tubes 2954 and 3053

Group No.	Energy, (a) MeV	A Priori Flux, (b) /cm ² .s	Adjusted Flux, /cm ² .s	Standard Deviation, %
1	1.492+001	1.216+009	1.220+009	52.
2	1.350+001	2.528+009	2.536+009	49.
3	1.162+001	1.172+010	1.176+010	45.
4	1.000+001	3.557+010	3.569+010	41.
5	8.607+000	9.000+010	9.030+010	37.
6	7.408+000	1.942+011	1.949+011	32.
7	6.065+000	5.382+011	5.400+011	27.
8	4.966+000	8.960+011	8.990+011	24.
9	3.679+000	2.071+012	2.078+012	20.
10	2.865+000	2.786+012	2.796+012	22.
11	2.231+000	3.678+012	3.690+012	27.
12	1.738+000	4.295+012	4.310+012	32.
13	1.353+000	4.448+012	4.463+012	37.
14	1.108+000	3.561+012	3.572+012	41.
15	8.208-001	5.767+012	5.786+012	45.
16	6.393-001	5.846+012	5.865+012	49.
17	4.979-001	5.461+012	5.478+012	52.
18	3.877-001	4.303+012	4.316+012	55.
19	3.020-001	4.239+012	4.252+012	57.
20	1.832-001	8.353+012	8.378+012	59.
21	1.111-001	7.187+012	7.208+012	61.
22	6.738-002	6.286+012	6.305+012	62.
23	4.087-002	5.881+012	5.898+012	63.
24	2.554-002	5.280+012	5.295+012	63.
25	1.989-002	2.774+012	2.782+012	64.
26	1.503-002	3.075+012	3.084+012	64.
27	9.119-003	5.092+012	5.106+012	64.
28	5.531-003	4.545+012	4.558+012	64.
29	3.355-003	5.148+012	5.163+012	65.
30	2.839-003	1.580+012	1.585+012	65.
31	2.404-003	1.567+012	1.572+012	65.
32	2.035-003	1.562+012	1.566+012	65.
33	1.234-003	4.676+012	4.689+012	65.
34	7.485-004	4.667+012	4.680+012	65.
35	4.540-004	4.587+012	4.600+012	65.
36	2.754-004	4.431+012	4.444+012	65.
37	1.670-004	4.377+012	4.389+012	65.
38	1.013-004	4.372+012	4.384+012	65.
39	6.144-005	4.334+012	4.346+012	65.
40	3.727-005	4.280+012	4.292+012	65.

(a) Lower Energy Bound

(b) Absolute calculated flux (normalized to the ⁵⁴Fe(n,p)⁵⁴Mn measured reaction rate).

TABLE 1. FERRET-SAND II Results for Pressure Tubes 2965 and 3053 (Cont'd)

Group No.	Energy, (a) MeV	A Priori Flux, (b) /cm ² .s	Adjusted Flux, /cm ² .s	Standard Deviation, %
41	2.260-005	4.166+012	4.178+012	65.
42	1.371-005	4.042+012	4.053+012	65.
43	8.315-006	3.937+012	3.948+012	65.
44	5.043-006	3.747+012	3.758+012	65.
45	3.059-006	3.795+012	3.806+012	65.
46	1.855-006	3.941+012	3.952+012	65.
47	1.125-006	4.064+012	4.076+012	65.
48	6.826-007	4.341+012	4.354+012	65.
49	4.140-007	5.581+012	5.597+012	65.
50	2.511-007	1.033+013	1.036+013	65.
51	1.523-007	1.840+013	1.845+013	65.
52	9.237-008	2.173+013	2.179+013	65.
53	1.000-010	3.740+013	3.751+013	65.
<hr/>				
FLUX, Total		2.677+014	2.685+014	43.
FLUX, E< 0.414 eV		8.786+013	8.811+013	61.
FLUX, E> 0.1 MeV		6.509+013	6.529+013	37.
FLUX, E> 1.0 MeV		2.458+013	2.466+013	21.
DPA/SECOND		3.850-008	3.863-008	24.
FLUENCE, (c) E> 1.0 MeV		8.866+021	8.896+021	21.
DPA(c)		1.389+001	1.393+001	24.

(a) Lower Energy Bound

(b) Absolute calculated flux (normalized to the ⁵⁴Fe(n,p)⁵⁴Mn measured reaction rate).

(c) Based on 3.607+008 effective full power seconds (EFPS) of exposure; units of /cm² for fluence and total number of displaced atoms of Iron per target atom of Iron for DPA.

Note: The EFPS value was determined by dividing the TIMWH derived tube power time-integrated exposure (Megawatt seconds) by an average reference tube power of 4.07 MW for Pressure Tube 3053.

TABLE 2. Comparison of Measured and Calculated Reaction Rates
used in the Analysis of Pressure Tubes 2954
(343.4 inches) and 3053 (351.2 inches)

Reaction	Cover	Reaction Rate, /s			Ratio Meas/Calc(a)	
		Meas(b)	A Priori Calc	Adj Calc	A Priori	Adj Calc
$^{54}\text{Fe}(n,p)^{54}\text{Mn}$	BARE	2.34E-12(8)	2.33E-12	2.34E-12	1.00	1.00

- (a) The relative WIMS 69 group input flux-spectrum was normalized to the $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ measured reaction rate so that the a priori ratio of meas/calc was unity.
- (b) The measured values of 2.23E-12 were multiplied by the ^{54}Fe target burnout correction factor of 1.048. The bracketed value is the one-sigma % uncertainty estimate.

TABLE 3. Estimated Fluences for Process Tube 2954

Tube Specimen Identification Number	Distance from End of Tube, Inches	Measured Reaction Rate, /s	Based on the Measured(a) $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ Reaction Rate		
			Fluence ($E > 0.1$ MeV), /cm ²	Fluence ($E > 0.18$ MeV), /cm ²	Fluence ($E > 1$ MeV), /cm ²
Fe 2B-2B	132.8	2.36E-13	2.49E+21	2.16E+21	9.41E+20
Fe F-2B	172.6	1.19E-12	1.26E+22	1.09E+22	4.75E+21
Fe 3F-2B	203.5	1.74E-12	1.83E+22	1.59E+22	6.92E+21
Fe 5F-2A1	229.8	2.11E-12	2.22E+22	1.93E+22	8.40E+21
Fe 7F-2C	264.0	2.18E-12	2.30E+22	2.00E+22	8.69E+21
Fe 2G-2A6	299.0	2.36E-12	2.49E+22	2.16E+22	9.39E+21
Fe 3G-3C	316.3	2.29E-12	2.42E+22	2.10E+22	9.14E+21
Fe J-4D2	343.4	2.23E-12	2.35E+22	2.04E+22	8.88E+21
Fe J-4B3A	360.2	2.14E-12	2.26E+22	1.96E+22	8.52E+21
Fe J-2A6B1	369.0	2.09E-12	2.20E+22	1.91E+22	8.32E+21
Fe 3J-2D	395.3	2.03E-12	2.14E+22	1.86E+22	8.08E+21
Fe 5J-4C	415.8	1.90E-12	2.01E+22	1.74E+22	7.57E+21
Fe 5J-2C	433.8	1.77E-12	1.87E+22	1.62E+22	7.05E+21
Fe 3K-2C	480.0	1.23E-12	1.30E+22	1.13E+22	4.91E+21
Fe 6K-4B	515.1	7.57E-12	7.99E+21	6.94E+21	3.02E+21
Fe 6K-2A	531.6	4.53E-13	4.78E+21	4.16E+21	1.81E+21

- (a) The measured values were increased by a factor of 1.048 to account for the burn-out of ^{54}Fe target atoms over the approximately 23 years of reactor operation; for the present study, however, no axial correction has been made for the drop-off of this central tube region burn-out factor with decreasing flux near and at the ends of the process tube. Consequently, the derived values of fluence at and near the ends of the process tubes are too high, but only by a very small percentage.

TABLE 4. Estimated Fluences for Process Tube 3053

Tube Specimen Identification Number	Distance from End of Tube, Inches	Measured Reaction Rate, /s	Based on the Measured(a) $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ Reaction Rate		
			Fluence ($E > 0.1$ MeV), /cm ²	Fluence ($E > 0.18$ MeV), /cm ²	Fluence ($E > 1$ MeV), /cm ²
Fe R-3C	151.7	6.10E-13	6.44E+21	5.60E+21	2.43E+21
Fe 3R-2C	183.0	1.14E-12	1.20E+22	1.05E+22	4.55E+21
Fe 5R-2A	213.0	1.82E-12	1.92E+22	1.67E+22	7.25E+21
Fe 7R-2A	231.6	2.00E-12	2.11E+22	1.84E+22	7.98E+21
Fe S-2C	250.0	2.12E-12	2.24E+22	1.95E+22	8.46E+21
Fe 3S-2C	283.9	2.17E-12	2.29E+22	1.99E+22	8.66E+21
Fe 5S-2B	312.8	2.21E-12	2.33E+22	2.03E+22	8.80E+21
Fe T-4B2B3	351.2	2.23E-12	2.35E+22	2.04E+22	8.86E+21
Fe T-4A6A	360.7	2.08E-12	2.19E+22	1.91E+22	8.28E+21
Fe 3T-2C	393.7	1.84E-12	1.94E+22	1.68E+22	7.32E+21
Fe 5T-3E	412.4	1.73E-12	1.83E+22	1.59E+22	6.91E+21
Fe X-2A	440.3	1.62E-12	1.71E+22	1.49E+22	6.47E+21
Fe 3X-1A	479.2	1.12E-12	1.18E+22	1.02E+22	4.45E+21
Fe 7X-2A	534.2	3.61E-13	3.81E+21	3.31E+21	1.44E+21

- (a) The measured values were increased by a factor of 1.048 to account for the burn-out of ^{54}Fe target atoms over the approximately 23 years of reactor operation; for the present study, however, no axial correction has been made for the drop-off of this central tube region burn-out factor with decreasing flux near and at the ends of the process tube. Consequently, the derived values of fluence at and near the ends of the process tubes are too high, but only by a very small percentage.

FIGURE 5. Pressure Tube 2954 Fluence Profiles

FIGURE 6. Pressure Tube 3053 Fluence Profiles

DISCUSSION

EXPOSURE PARAMETER VALUES AND UNCERTAINTIES

The counting statistics associated with the measurement of the ^{54}Mn and ^{60}Co specific activities is estimated to be in the ± 3 to 5% (one sigma) range. The range for ^{58}Co is ± 8 to 10%. The estimated uncertainties associated with the chemistry measurements of the elemental content of Zr, Fe, Ni, Co and Sn are discussed in Appendix B. The chemistry uncertainty values for Fe and Ni are estimated to be $\pm 6\%$ for each element. This results in combined uncertainties of approximately $\pm 8\%$ and $\pm 12\%$, respectively, for the measured reaction rates for the $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ and $^{58}\text{Ni}(n,p)^{58}\text{Co}$ reactions.

The uncertainty associated with the determination of the time-history saturation correction factor for the ^{54}Mn and other reaction products was not investigated and is not included in the $\pm 8\%$ and $\pm 12\%$ values established for the iron and nickel reaction rates. The uncertainties associated with making the burn-in and burn-out corrections have not been determined. Additional study will be required to address these and other reaction rate uncertainty issues of concern. As noted in Appendix G, a N Reactor pressure tube physics-dosimetry experiment would be required to establish the exact biases and uncertainties associated with the derived $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ spectral-averaged cross-section. The current values and uncertainties associated with the FERRET-SAND II derived exposure and exposure rates are likely, therefore, to be subject to some change.

The results of the present, as well as other studies (Lippincott et al. 1987; Kellogg et al. 1987; Anderson and Lippincott 1986; McElroy 1985), support the on-going need to use dosimetry adjusted reactor physics calculations to determine reliable (± 5 to $\pm 20\%$, one standard deviation) exposure parameter values for test, production, and power reactor surveillance programs.

COMPARISON WITH PREVIOUS RESULTS

Information on previous physics-dosimetry results for Pressure

Tubes 1165 (Marshall and Scott 1984), and 1054 and 2755 (Lawrence and Trimble 1985) was briefly reviewed as a part of the present study. It was noted that the fast ($E > 1$ MeV) flux and fluence values which were derived by Marshall and Scott (1984) are based on the use of a fission spectrum averaged $\bar{\sigma}^f(E > 1 \text{ MeV})$ $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ cross-section value (not given); this procedure can be expected to yield exposure and exposure rate values for Pressure Tube 1165 that are biased low from those reported herein for Pressure Tubes 2954 and 3053.

This bias can be estimated to be equal to the ratio (1.23) of the $^{54}\text{Fe}(n,p)^{54}\text{Mn}$ reaction's $\bar{\sigma}^f(E > 1 \text{ MeV})$ value of 117 mb, (a) divided by the N Reactor's calculated spectral averaged ($E > 1$ MeV) cross section value of 94.9 mb (based on the WIMS 69 group flux-spectrum and the ENDF/B-V iron cross section).

When combined with a 1.048 correction factor for the burn-out of ^{54}Fe target atoms, the total bias between the Pressure Tube 1165 and 2954 and 3053 results would appear to be about a factor of 1.30. A more careful and detailed study was recommended to determine the actual bias factor; such a study has been completed and the results, reported in Appendix G, show that a factor of 1.30 is correct for the neutron flux values. However, the fluence values needed a correction factor of 1.36; because of the large differences, a study of the existence of any bias between the 1985 (Lawrence and Trimble, Pressure Tubes 2755 and 1054) and the present results should be considered.

A POSTERIORI SURVEILLANCE DOSIMETRY SYSTEM

The present (preliminary) analysis indicates that a reliable a posteriori pressure tube surveillance dosimetry system [two high-energy $\{^{54}\text{Fe}(n,p)^{54}\text{Mn}$ and $^{58}\text{Ni}(n,p)^{58}\text{Co}\}$ and three low-energy $\{^{94}\text{Zr}(n,\gamma)^{95}\text{Zr}$, $^{124}\text{Sn}(n,\gamma)^{125}\text{Sn}$, and $^{59}\text{Co}(n,\gamma)^{60}\text{Co}\}$ reactions] could be established and validated for adjusting past, present and future WIMS and MCNP calculations. This would add significant value to existing N Reactor surveillance program efforts because there is a need for having low-energy (thermal and inter-

(a) The value of $\bar{\sigma}^f(E > 1 \text{ MeV})$ is obtained from the values of $\bar{\sigma}^f = 80 \text{ mb}/0.6917$ (the fraction of the fission spectrum above 1 MeV); see, for example, ASTM 1987a.

mediate neutron energy regions), as well as high-energy neutron exposure and exposure rate parameter values.

The low energy data are needed in the assessment of radiation damage to components and materials, for calculating helium production from boron and other impurities, for gamma heating and thermal hydraulics analysis, for the validation of reactor physic calculations and for making burn-in and burn-out corrections for dosimetry sensor reactions.

In regards to the establishment of an a posteriori dosimetry system for N Reactor surveillance programs, the FERRET-SAND II solution results are compared in Table 5 with interpolated results (provided by Morgan, Appendix C) from absolute MCNP calculations. The agreement is excellent and is much better than would normally be expected. Further study is needed to verify that the agreement is as good as indicated by this preliminary comparison.

TABLE 5. Comparison of MCNP Calculated and FERRET-SAND II Results for Pressure Tubes 2954 and 3053

<u>Integragal Flux and Fluence Values</u>	<u>MCNP(a) Calculation, /cm².s</u>	<u>FERRET-SAND II Adjusted Flux, /cm².s</u>	<u>FERRET-SAND II</u>
FLUX, E >0.0 MeV	2.58E+014	2.69E+014	43. %
FLUX, E <0.414 eV	8.75E+013	8.81E+013	61. %
FLUX, E >0.1 MeV	6.30E+013	6.53E+013	37. %
FLUX, E >0.18 MeV	5.53E+013	5.68E+013	< 37. %
FLUX, E >1.0 MeV	2.35E+013	2.47E+013	21. %
FLUENCE, (b) E >1.0 MeV	8.48E+021	8.90E+021	21. %

(a) Morgan's interpolated values based on absolute MCNP calculations; details are discussed in Appendix C.

(b) Based on 3.607+008 effective full power seconds of exposure; units of /cm².

RECOMMENDATIONS

The following recommendations are made by the PNL-NDC staff to help improve the quality of the physics-dosimetry needed to support the DOE-WHC N Reactor Surveillance Program:

1. Previous physics-dosimetry results for Pressure Tubes 1165 (Marshall and Scott 1984), 1054 and 2755 (Lawrence and Trimble 1985) should be re-evaluated and updated so that they are self-consistent with the presently derived FERRET-SAND II exposure parameter values for Pressure Tube 2954 and Pressure Tube 3053.
2. Additional work should be done on the calculations and associated uncertainties of the burn-in and burn-out correction factors for the different isotopes of interest. The determination of such correction factors has not been addressed in previous studies, and in the present study, the drop-off in the burn-out of ^{54}Fe target atoms near and at the ends of the pressure tubes (as the thermal flux level decreases) has not been evaluated.
3. The reason for the anomaly in the measured cobalt impurity concentrations should be investigated.
4. Consideration should be given to the completion of the development and verification of an improved a posteriori pressure tube surveillance dosimetry system [of two high-energy $\{^{54}\text{Fe}(n,p)^{54}\text{Mn}$ and $^{58}\text{Ni}(n,p)^{58}\text{Co}\}$ and three low-energy $\{^{94}\text{Zr}(n,\gamma)^{95}\text{Zr}$, $^{124}\text{Sn}(n,\gamma)^{125}\text{Sn}$, and $^{59}\text{Co}(n,\gamma)^{60}\text{Co}\}$ reactions] that can be routinely used for FERRET-SAND II adjustments of WIMS and MCNP calculations for N Reactor pressure tubes. There is an on-going need to have reliable low-energy (thermal and intermediate neutron energy region), as well as high-energy neutron exposure and exposure rate parameter data, for the assessment of radiation damage to components and materials. This same information is needed for the derivation and subsequent application of physics-dosimetry results.

5. The ENDF/B-V energy dependent cross-sections for the $^{94}\text{Zr}(n,\gamma)^{95}\text{Zr}$ and $^{124}\text{Sn}(n,\gamma)^{125}\text{Sn}$ reactions should be processed for addition to the SAND II 620 group (SIGS) and the FERRET 53 group (LIB) dosimetry files for use in N Reactor pressure tube physics-dosimetry studies.
6. It is suggested that consideration be given to the analyses of pressure tube specimens for helium and boron content to see if (n,He) stable product reaction results can be used to provide an additional and independent check on the reliability of existing and future physics-dosimetry results. Helium reaction (n,He) results provide a direct measurement of time-integrated exposure parameter values (ASTM 1987c) without the need for establishing a flux-level time history for a pressure tube.

Based on LWR pressure vessel physics-dosimetry-metallurgy results (McElroy et al. 1986a, Gold and McElroy 1986, McElroy et al. 1986b, Simons 1986, McElroy and Gold 1987, Lippincott et al. 1987, Kellogg et al. 1987) knowledge of the helium content of the pressure tubes should be available to help in the assessment of the role that neutron-induced helium production might have in contributing to irradiation-induced physical and mechanical property changes in N Reactor pressure tubes.

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APPENDIX A

TIMWH INPUT DATA AND DERIVED REACTION RATES
FOR N-REACTOR PRESSURE TUBES 2954 AND 3053

APPENDIX A

TIMWH INPUT DATA AND DERIVED REACTION RATES FOR PRESSURE TUBE 2954 DOSIMETRY

DATE AND TIME IN - 2/14/64 AT 23:59
DATE AND TIME OUT - 1/ 7/87 AT 8:57

EXPOSURE- MW SEC. MW HOURS MWD
1.4088+009 3.9134+005 1.6306+004

TOTAL TIME IN REACTOR = 200696.9609 HRS
NORMALIZED TO POWER OF 4.07 MW

POWER (MW) WAS-

3.0740	.0000	3.0740	.0000	3.0740
.0000	3.0740	.0000	3.0740	.0000
3.8620	.0000	3.8620	.0000	3.8620
.0000	3.8620	.0000	3.8620	.0000
3.8990	.0000	3.8990	.0000	3.8990
.0000	3.8990	.0000	3.4930	.0000
3.4930	.0000	3.9100	.0000	3.9100
.0000	3.9240	.0000	3.9240	.0000
3.9240	.0000	3.9240	.0000	3.8310
.0000	3.8310	.0000	3.8310	.0000
3.6190	.0000	3.6190	.0000	3.6190
.0000	3.6190	.0000	3.6190	.0000
3.6190	.0000	3.6190	.0000	3.5480
.0000	3.5480	.0000	3.5480	.0000
3.5480	.0000	3.8610	.0000	3.8610
.0000	3.8610	.0000	4.3690	.0000
4.3690	.0000	4.3690	.0000	4.3240
.0000	4.3240	.0000	4.3240	.0000
4.3060	.0000	4.3060	.0000	4.3060
.0000	4.2520	.0000	4.2520	.0000
4.2520	.0000	4.0940	.0000	4.0940
.0000	4.0940	.0000	4.0940	.0000
4.0940	.0000	3.8160	.0000	3.8160
.0000	3.9730	.0000	3.9730	.0000
3.7130	.0000	3.7130	.0000	3.7130
.0000	3.7130	.0000	3.6330	.0000
3.6330	.0000	3.8040	.0000	3.8040
.0000	3.8040	.0000	3.9710	.0000
3.9710	.0000	3.8320	.0000	3.8320
.0000	3.8320	.0000	3.7820	2.5040
3.7820	.0000	2.7380	.0000	.8770
3.7820	1.4860	3.7820	.0000	3.7820
.0000	3.7820	.0000	3.7820	.0000
4.3690	1.5140	4.2510	2.2320	.0000
2.5200	4.2500	.0000	.8550	4.5470
1.7120	4.6080	2.4330	4.3840	

TIME (HRS) WAS-

720.0000	2952.0000	1682.0000	1992.0000	3000.0000
2867.0000	947.0000	689.0000	871.0000	696.0000
1784.0000	1320.0000	1152.0000	1190.0000	818.0000
480.0000	1489.0000	463.0000	1000.0000	488.0000
1216.0000	452.0000	628.0000	1831.0000	856.0000
1969.0000	1248.0000	297.0000	1503.0000	528.0000
912.0000	1562.0000	2902.0000	912.0000	1296.0000
245.0000	883.0000	1583.0000	2521.0000	1080.0000
912.0000	576.0000	840.0000	410.0000	550.0000
1533.0000	1755.0000	840.0000	1896.0000	6696.0000
1805.0000	2299.0000	528.0000	2496.0000	720.0000
185.0000	1207.0000	491.0000	2533.0000	1532.0000
724.0000	953.0000	1999.0000	356.0000	1492.0000
490.0000	2178.0000	1608.0000	2108.0000	936.0000
1248.0000	435.0000	2877.0000	1608.0000	1680.0000
1896.0000	1416.0000	1451.0000	1693.0000	2280.0000
1680.0000	2016.0000	1632.0000	1447.0000	2177.0000
888.0000	1200.0000	744.0000	840.0000	451.0000
1637.0000	2040.0000	1872.0000	1056.0000	1680.0000
435.0000	1725.0000	1800.0000	2520.0000	1944.0000
624.0000	441.0000	1420.0000	7224.0000	618.0000
2161.0000	1865.0000	2399.0000	840.0000	232.0000
632.0000	227.0000	685.0000	2738.0000	1366.0000
379.0000	677.0000	841.0000	1967.0000	227.0000
733.0000	1344.0000	624.0000	432.0000	600.0000
901.0000	1139.0000	807.0000	1449.0000	521.0000
607.0000	1752.0000	862.0000	506.0000	984.0000
622.0000	890.0000	526.0000	1298.0000	1111.0000
713.0000	357.0000	867.0000	260.0000	748.0000
336.0000	528.0000	3309.3300	209.9900	248.7000
218.7300	141.3000	288.0000	312.0000	168.0000
96.0000	72.0000	764.5600	317.6100	57.3300
109.7000	84.8200	89.2000	199.9200	961.8900
295.0000	72.0000	408.0000	96.0000	192.0000
168.0000	275.1000	1308.9000	120.0000	264.0000
72.0000	408.0000	96.0000	152.9000	

I	REACTION NAME	DECAY CONST. (Sec-1)	FISSION YLD FRACTION	SATURATION FACTOR	CORRECTION TO TOR	K FACTOR(1.0E-19)* (mg/atom)
1	FE54(N,P)MMS4	2.567-008	1.000+000	2.177+000	1.000+000	1.599+001
2	N158(N,P)CO58	1.132-007	1.000+000	1.845+000	1.000+000	1.427+000
3	CU63(N,A)CO60	4.170-009	1.000+000	2.224+000	1.000+000	1.527+000
4	TI46(N,P)TI46	9.568-008	1.000+000	1.914+000	1.000+000	9.955+000
5	CO59(N,G)CO60	4.167-009	1.000+000	2.224+000	1.000+000	9.786-001
6	FE58(N,G)FE59	1.799-007	1.000+000	1.636+000	1.000+000	2.810+002
7	U238(N,F)CS137	7.279-010	6.000-002	5.106+000	1.000+000	3.953+000
8	U238(N,F)ZR95	1.246-007	5.105-002	1.804+000	1.000+000	3.953+000
9	U238(N,F)RU103	2.035-007	6.229-002	1.577+000	1.000+000	3.953+000
10	U238(N,F)CE141	2.468-007	5.456-002	1.482+000	1.000+000	3.953+000
11	U238(N,F)RU106	2.180-008	2.513-002	2.179+000	1.000+000	3.953+000
12	NP237(N,F)CS137	7.279-010	6.267-002	5.106+000	1.000+000	3.936+000
13	NP237(N,F)ZR95	1.246-007	5.699-002	1.804+000	1.000+000	3.936+000
14	NP237(N,F)RU103	2.035-007	5.584-002	1.577+000	1.000+000	3.936+000
15	NP237(N,F)CE141	2.468-007	5.439-002	1.482+000	1.000+000	3.936+000
16	NP237(N,F)RU106	2.180-008	2.240-002	2.179+000	1.000+000	3.936+000
17	NP237(N,F)CE144	2.821-008	4.198-002	2.174+000	1.000+000	3.936+000
18	ZR94(N,G)ZR95	1.252-007	1.000+000	1.802+000	1.000+000	8.706+000
19	SN124(N,G)SN125	7.935-009	1.000+000	2.158+000	1.000+000	3.520+001
20	U238(N,F)CE144	2.821-008	4.543-002	2.174+000	1.000+000	3.953+000

* UNIT CONVERSION FACTOR FROM DPS/MG TO DPS/ATOM

REACTION NAME	IDENT	EOI RATE* (DPS/NUCL)	SATURATION FACTOR	GRADIENT FACTOR	IMPURITY FACTOR	REACTION RATE (RPS/NUCL)	% UNCERT (1 STD)
ZR94(N,G)ZR95	Zr 2B-2B	3.34-013	1.802+000	1.000	1.000	6.01-013	5.0**
	Zr F-2B	1.42-012	1.802+000	1.000	1.000	2.55-012	
	Zr 3F-2B	2.15-012	1.802+000	1.000	1.000	3.87-012	
	Zr 5F-2A1	2.35-012	1.802+000	1.000	1.000	4.24-012	
	Zr 7F-2C	2.53-012	1.802+000	1.000	1.000	4.57-012	
	Zr 2G-2A6	2.59-012	1.802+000	1.000	1.000	4.66-012	
	Zr 3G-3C	2.54-012	1.802+000	1.000	1.000	4.58-012	
	Zr J-402	2.27-012	1.802+000	1.000	1.000	4.09-012	
	Zr J-4B3A	2.32-012	1.802+000	1.000	1.000	4.17-012	
	Zr J-2A6B1	2.39-012	1.802+000	1.000	1.000	4.30-012	
	Zr 3J-2D	2.40-012	1.802+000	1.000	1.000	4.33-012	
	Zr 5J-4C	2.29-012	1.802+000	1.000	1.000	4.12-012	
	Zr 5J-2C	2.10-012	1.802+000	1.000	1.000	3.78-012	
	Zr 3K-2C	1.49-012	1.802+000	1.000	1.000	2.69-012	
	Zr 6K-4B	8.98-013	1.802+000	1.000	1.000	1.62-012	
	Zr 6K-2A	5.65-013	1.802+000	1.000	1.000	1.02-012	

CO59(N,G)CO60	Co 2B-2B	9.27-013	2.224+000	1.000	1.000	2.06-012	TBD
	Co F-2B	3.14-012	2.224+000	1.000	1.000	6.98-012	
	Co 3F-2B	4.76-012	2.224+000	1.000	1.000	1.06-011	
	Co 5F-2A1	5.28-012	2.224+000	1.000	1.000	1.17-011	
	Co 7F-2C	6.20-012	2.224+000	1.000	1.000	1.38-011	
	Co 2G-2A6	6.34-012	2.224+000	1.000	1.000	1.41-011	
	Co 3G-3C	6.12-012	2.224+000	1.000	1.000	1.36-011	
	Co J-402	5.01-012	2.224+000	1.000	1.000	1.11-011	
	Co J-483A	5.09-012	2.224+000	1.000	1.000	1.13-011	
	Co J-2A681	5.40-012	2.224+000	1.000	1.000	1.20-011	
	Co 3J-2D	5.68-012	2.224+000	1.000	1.000	1.26-011	
	Co 5J-4C	5.39-012	2.224+000	1.000	1.000	1.20-011	
	Co 5J-2C	4.93-012	2.224+000	1.000	1.000	1.10-011	
	Co 3K-2C	3.44-012	2.224+000	1.000	1.000	7.66-012	
	Co 6K-4B	2.07-012	2.224+000	1.000	1.000	4.60-012	
	Co 6K-2A	1.34-012	2.224+000	1.000	1.000	2.98-012	
SN124(N,G)SN125	Sn 2B-2B	3.41-012	2.158+000	1.000	1.000	7.36-012	5.8
	Sn F-2B	1.45-011	2.158+000	1.000	1.000	3.13-011	
	Sn 3F-2B	2.09-011	2.158+000	1.000	1.000	4.52-011	
	Sn 5F-2A1	2.34-011	2.158+000	1.000	1.000	5.05-011	
	Sn 7F-2C	2.56-011	2.158+000	1.000	1.000	5.52-011	
	Sn 2G-2A6	2.56-011	2.158+000	1.000	1.000	5.52-011	
	Sn 3G-3C	2.52-011	2.158+000	1.000	1.000	5.43-011	
	Sn J-402	2.37-011	2.158+000	1.000	1.000	5.10-011	
	Sn J-483A	2.38-011	2.158+000	1.000	1.000	5.14-011	
	Sn J-2A681	2.41-011	2.158+000	1.000	1.000	5.19-011	
	Sn 3J-2D	2.39-011	2.158+000	1.000	1.000	5.17-011	
	Sn 5J-4C	2.26-011	2.158+000	1.000	1.000	4.87-011	
	Sn 5J-2C	2.14-011	2.158+000	1.000	1.000	4.62-011	
	Sn 3K-2C	1.52-011	2.158+000	1.000	1.000	3.29-011	
	Sn 6K-4B	9.82-012	2.158+000	1.000	1.000	2.12-011	
	Sn 6K-2A	6.70-012	2.158+000	1.000	1.000	1.45-011	
FES4(N,P)MN54	Fe 2B-2B	1.08-013	2.177+000	1.000	1.000	2.36-013	7.8
	Fe F-2B	5.48-013	2.177+000	1.000	1.000	1.19-012	
	Fe 3F-2B	7.98-013	2.177+000	1.000	1.000	1.74-012	
	Fe 5F-2A1	9.69-013	2.177+000	1.000	1.000	2.11-012	
	Fe 7F-2C	1.00-012	2.177+000	1.000	1.000	2.18-012	
	Fe 2G-2A6	1.08-012	2.177+000	1.000	1.000	2.36-012	
	Fe 3G-3C	1.05-012	2.177+000	1.000	1.000	2.29-012	
	Fe J-402	1.02-012	2.177+000	1.000	1.000	2.23-012	
	Fe J-483A	9.82-013	2.177+000	1.000	1.000	2.14-012	
	Fe J-2A681	9.59-013	2.177+000	1.000	1.000	2.09-012	
	Fe 3J-2D	9.32-013	2.177+000	1.000	1.000	2.03-012	
	Fe 5J-4C	8.73-013	2.177+000	1.000	1.000	1.90-012	
	Fe 5J-2C	8.13-013	2.177+000	1.000	1.000	1.77-012	
	Fe 3K-2C	5.66-013	2.177+000	1.000	1.000	1.23-012	
	Fe 6K-4B	3.48-013	2.177+000	1.000	1.000	7.57-013	
	Fe 6K-2A	2.08-013	2.177+000	1.000	1.000	4.53-013	

Ni58(N,P)Co58	Ni 2B-2B	1.47-013	1.845+000	1.000	1.000	2.72-013	12.
	Ni F-2B	6.78-013	1.845+000	1.000	1.000	1.25-012	
	Ni 3F-2B	8.87-013	1.845+000	1.000	1.000	1.64-012	
	Ni 5F-2A1	1.04-012	1.845+000	1.000	1.000	1.91-012	
	Ni 7F-2C	9.28-013	1.845+000	1.000	1.000	1.71-012	
	Ni 2G-2A6	1.01-012	1.845+000	1.000	1.000	1.86-012	
	Ni 3G-3C	9.52-013	1.845+000	1.000	1.000	1.76-012	
	Ni J-402	1.04-012	1.845+000	1.000	1.000	1.93-012	
	Ni J-483A	1.05-012	1.845+000	1.000	1.000	1.94-012	
	Ni J-2A681	1.02-012	1.845+000	1.000	1.000	1.88-012	
	Ni 3J-2D	9.42-013	1.845+000	1.000	1.000	1.74-012	
	Ni 5J-4C	8.85-013	1.845+000	1.000	1.000	1.63-012	
	Ni 5J-2C	9.47-013	1.845+000	1.000	1.000	1.75-012	
	Ni 3K-2C	6.74-013	1.845+000	1.000	1.000	1.24-012	
	Ni 6K-4B	4.62-013	1.845+000	1.000	1.000	8.53-013	
	Ni 6K-2A	2.58-013	1.845+000	1.000	1.000	4.76-013	

* End-of-irradiation (EOI) specific activity.

** The counting statistics are estimated by Matsumoto to be in the range of + 3 to 5% (one sigma) for the Mn54, Co59, Zr95, and Sn125 and + 8 to 10% for the Co58. The uncertainties (EOI) in the measured content (weight) of Fe54, Zr94, Sn124, and Ni58 are estimated by Matsumoto to be + 6%, + 0.5%, + 3% and + 6%, respectively. An EOI value for Co59 was not determined; see Appendix B. The counting statistics and isotopic uncertainties were combined in quadrature to obtain the one sigma measured reaction rate uncertainties. Uncertainties associated with target atom and reaction product atom burn-in and burn-out correction factors are not included; see report text and Appendix C. The specified uncertainty applies for for all measured values for each set of reaction rates; e.g., all Zr94(n,g)Zr95 specimens Zr 2B-2B thru Zr 6K-2A.

APPENDIX A (Cont'd)

TIMWH INPUT DATA AND DERIVED REACTION RATES FOR PRESSURE TUBE 3053 DOSIMETRY

DATE AND TIME IN - 2/14/64 AT 23:59

EXPOSURE- MW SEC. MW HOURS MWD

DATE AND TIME OUT- 1/ 7/87 AT 8:57

1.4681+009 4.0781+005 1.6992+004

TOTAL TIME IN REACTOR = 200696.9609 HRS
NORMALIZED TO POWER OF 4.07 MW

POWER (MW) WAS-

2.9200	.0000	2.9200	.0000	2.9200
.0000	2.9200	.0000	3.8140	.0000
3.8140	.0000	3.8140	.0000	3.8140
.0000	4.2880	.0000	4.2880	.0000
4.2880	.0000	4.1910	.0000	4.1090
.0000	4.1090	.0000	3.6360	.0000
3.6370	.0000	4.2480	.0000	4.2480
.0000	4.2480	.0000	4.2480	.0000
4.2480	.0000	4.2480	.0000	4.2480
.0000	4.1610	.0000	4.1610	.0000
3.8390	.0000	3.8390	.0000	3.8390
.0000	4.1660	.0000	4.1660	.0000
4.2210	.0000	4.2210	.0000	4.2210
.0000	3.6930	.0000	3.6920	.0000
3.9620	.0000	3.9200	.0000	3.9200
.0000	3.9200	.0000	4.3890	.0000
4.3890	.0000	4.3890	.0000	4.3360
.0000	4.3360	.0000	4.3360	.0000
4.2620	.0000	4.2620	.0000	4.2620
.0000	4.2320	.0000	4.2320	.0000
4.2320	.0000	4.0420	.0000	4.0420
.0000	4.0420	.0000	4.0420	.0000
4.0210	.0000	4.0210	.0000	4.1000
.0000	4.1000	.0000	4.2060	.0000
4.0630	.0000	4.0630	.0000	4.0630
.0000	3.8050	.0000	3.8050	.0000
3.6960	.0000	3.6960	.0000	3.6960
.0000	4.1490	.0000	4.1490	.0000
3.9210	.0000	3.9210	.0000	4.0220
.0000	4.0220	.0000	3.9270	2.6000
3.9270	.0000	2.8430	.0000	.9110
3.9270	1.5430	3.9270	.0000	4.0820
.0000	4.0820	.0000	4.0820	.0000
4.5810	1.5670	4.2970	2.2560	.0000
2.7020	4.5570	.0000	.9010	4.7940
1.8130	4.9020	2.5620	4.6170	

TIME (HRS) WAS-

720.0000	2952.0000	1682.0000	1992.0000	3000.0000
2867.0000	947.0000	689.0000	871.0000	696.0000
1784.0000	1320.0000	1152.0000	1190.0000	818.0000
480.0000	1489.0000	463.0000	1000.0000	488.0000
1216.0000	452.0000	628.0000	1831.0000	856.0000
1969.0000	1248.0000	297.0000	1503.0000	528.0000
912.0000	1562.0000	2902.0000	912.0000	1296.0000
245.0000	883.0000	1583.0000	2521.0000	1080.0000
912.0000	576.0000	840.0000	410.0000	550.0000
1533.0000	1755.0000	840.0000	1896.0000	6696.0000
1805.0000	2299.0000	528.0000	2496.0000	720.0000
185.0000	1207.0000	491.0000	2533.0000	1532.0000
724.0000	953.0000	1999.0000	356.0000	1492.0000
490.0000	2178.0000	1608.0000	2108.0000	936.0000
1248.0000	435.0000	2877.0000	1608.0000	1680.0000
1896.0000	1416.0000	1451.0000	1693.0000	2280.0000
1680.0000	2016.0000	1632.0000	1447.0000	2177.0000
888.0000	1200.0000	744.0000	840.0000	451.0000
1637.0000	2040.0000	1872.0000	1056.0000	1680.0000
435.0000	1725.0000	1800.0000	2520.0000	1944.0000
624.0000	441.0000	1420.0000	7224.0000	618.0000
2161.0000	1865.0000	2399.0000	840.0000	232.0000
632.0000	227.0000	685.0000	2738.0000	1366.0000
379.0000	677.0000	841.0000	1967.0000	227.0000
733.0000	1344.0000	624.0000	432.0000	600.0000
901.0000	1139.0000	807.0000	1449.0000	521.0000
607.0000	1752.0000	862.0000	506.0000	984.0000
622.0000	890.0000	526.0000	1298.0000	1111.0000
713.0000	357.0000	867.0000	260.0000	748.0000
336.0000	528.0000	3309.3300	209.9900	248.7000
218.7300	141.3000	288.0000	312.0000	168.0000
96.0000	72.0000	764.5600	317.6100	57.3300
109.7000	84.8200	89.2000	199.9200	961.8900
295.0000	72.0000	408.0000	96.0000	192.0000
168.0000	275.1000	1308.9000	120.0000	264.0000
72.0000	408.0000	96.0000	152.9000	

I	REACTION NAME	DECAY CONST. (Sec-1)	FISSION YLD FRACTION	SATURATION FACTOR	CORRECTION TO TOR	K FACTOR(1.0E-19)* (mg/atom)
1	FE54(N,P)MH54	2.567-008	1.000+000	2.089+000	1.000+000	1.599+001
2	NI58(N,P)CO58	1.132-007	1.000+000	1.751+000	1.000+000	1.427+000
3	CU63(N,A)CO60	4.170-009	1.000+000	2.150+000	1.000+000	1.527+000
4	TI46(N,P)TI46	9.568-008	1.000+000	1.818+000	1.000+000	9.955+000
5	CO59(N,G)CO60	4.167-009	1.000+000	2.150+000	1.000+000	9.786-001
6	FE58(N,G)FE59	1.799-007	1.000+000	1.549+000	1.000+000	2.810+002
7	U238(N,F)CS137	7.279-010	6.000-002	4.907+000	1.000+000	3.953+000
8	U238(N,F)ZR95	1.246-007	5.105-002	1.711+000	1.000+000	3.953+000
9	U238(N,F)RU103	2.035-007	6.229-002	1.492+000	1.000+000	3.953+000
10	U238(N,F)CE141	2.468-007	5.456-002	1.402+000	1.000+000	3.953+000
11	U238(N,F)RU106	2.180-008	2.513-002	2.093+000	1.000+000	3.953+000
12	NP237(N,F)CS137	7.279-010	6.267-002	4.907+000	1.000+000	3.936+000
13	NP237(N,F)ZR95	1.246-007	5.699-002	1.711+000	1.000+000	3.936+000
14	NP237(N,F)RU103	2.035-007	5.584-002	1.492+000	1.000+000	3.936+000
15	NP237(N,F)CE141	2.468-007	5.439-002	1.402+000	1.000+000	3.936+000
16	NP237(N,F)RU106	2.180-008	2.240-002	2.093+000	1.000+000	3.936+000
17	NP237(N,F)CE144	2.821-008	4.198-002	2.085+000	1.000+000	3.936+000
18	ZR94(N,G)ZR95	1.252-007	1.000+000	1.709+000	1.000+000	8.706+000
19	SN124(N,G)SN125	7.935-009	1.000+000	2.089+000	1.000+000	3.520+001
20	U238(N,F)CE144	2.821-008	4.543-002	2.085+000	1.000+000	3.953+000

* UNIT CONVERSION FACTOR FROM DPS/MG TO DPS/ATOM

REACTION NAME	IDENT	EOI RATE* (DPS/NUCL)	SATURATION FACTOR	GRADIENT FACTOR	IMPURITY FACTOR	REACTION RATE (RPS/NUCL)	% UNCERT (1 STD)
ZR94(N,G)ZR95	Zr R-3C	7.18-013	1.709+000	1.000	1.000	1.23-012	5.0**
	Zr 3R-2C	1.61-012	1.709+000	1.000	1.000	2.75-012	
	Zr 5R-2A	2.20-012	1.709+000	1.000	1.000	3.75-012	
	Zr 7R-2A	2.40-012	1.709+000	1.000	1.000	4.11-012	
	Zr S-2C	2.52-012	1.709+000	1.000	1.000	4.31-012	
	Zr 3S-2C	2.60-012	1.709+000	1.000	1.000	4.44-012	
	Zr 5S-2B	2.57-012	1.709+000	1.000	1.000	4.40-012	
	Zr T-4B2B3	2.24-012	1.709+000	1.000	1.000	3.83-012	
	Zr T-4A6A	2.27-012	1.709+000	1.000	1.000	3.89-012	
	Zr 3T-2C	2.40-012	1.709+000	1.000	1.000	4.11-012	
	Zr 5T-3E	2.29-012	1.709+000	1.000	1.000	3.91-012	
	Zr X-2A	1.96-012	1.709+000	1.000	1.000	3.35-012	
	Zr 3X-1A	1.46-012	1.709+000	1.000	1.000	2.49-012	
	Zr 7X-2A	4.62-013	1.709+000	1.000	1.000	7.89-013	

CO59(N,G)CO60	Co R-3C	2.90-011	2.150+000	1.000	1.000	6.25-011	TBD
	Co 3R-2C	5.56-011	2.150+000	1.000	1.000	1.20-010	
	Co 5R-2A	6.82-011	2.150+000	1.000	1.000	1.47-010	
	Co 7R-2A	7.21-011	2.150+000	1.000	1.000	1.55-010	
	Co S-2C	7.35-011	2.150+000	1.000	1.000	1.58-010	
	Co 3S-2C	7.55-011	2.150+000	1.000	1.000	1.62-010	
	Co 5S-2B	7.24-011	2.150+000	1.000	1.000	1.56-010	
	Co T-4B2B3	6.38-011	2.150+000	1.000	1.000	1.37-010	
	Co T-4A6A	6.54-011	2.150+000	1.000	1.000	1.41-010	
	Co 3T-2C	6.77-011	2.150+000	1.000	1.000	1.46-010	
	Co 5T-3E	6.48-011	2.150+000	1.000	1.000	1.39-010	
	Co X-2A	5.72-011	2.150+000	1.000	1.000	1.23-010	
	Co 3X-1A	4.45-011	2.150+000	1.000	1.000	9.57-011	
	Co 7X-2A	1.71-011	2.150+000	1.000	1.000	3.68-011	
SN124(N,G)SN125	Sn R-3C	8.39-012	2.089+000	1.000	1.000	1.75-011	5.8
	Sn 3R-2C	1.64-011	2.089+000	1.000	1.000	3.42-011	
	Sn 5R-2A	2.24-011	2.089+000	1.000	1.000	4.68-011	
	Sn 7R-2A	2.37-011	2.089+000	1.000	1.000	4.94-011	
	Sn S-2C	2.46-011	2.089+000	1.000	1.000	5.15-011	
	Sn 3S-2C	2.59-011	2.089+000	1.000	1.000	5.40-011	
	Sn 5S-2B	2.52-011	2.089+000	1.000	1.000	5.26-011	
	Sn T-4B2B3	2.32-011	2.089+000	1.000	1.000	4.85-011	
	Sn T-4A6A	2.35-011	2.089+000	1.000	1.000	4.91-011	
	Sn 3T-2C	2.37-011	2.089+000	1.000	1.000	4.94-011	
	Sn 5T-3E	2.25-011	2.089+000	1.000	1.000	4.70-011	
	Sn X-2A	2.00-011	2.089+000	1.000	1.000	4.19-011	
	Sn 3X-1A	1.50-011	2.089+000	1.000	1.000	3.14-011	
	Sn 7X-2A	5.67-012	2.089+000	1.000	1.000	1.18-011	
FE54(N,P)MN54	Fe R-3C	2.92-013	2.089+000	1.000	1.000	6.10-013	7.8
	Fe 3R-2C	5.47-013	2.089+000	1.000	1.000	1.14-012	
	Fe 5R-2A	8.72-013	2.089+000	1.000	1.000	1.82-012	
	Fe 7R-2A	9.59-013	2.089+000	1.000	1.000	2.00-012	
	Fe S-2C	1.02-012	2.089+000	1.000	1.000	2.12-012	
	Fe 3S-2C	1.04-012	2.089+000	1.000	1.000	2.17-012	
	Fe 5S-2B	1.06-012	2.089+000	1.000	1.000	2.21-012	
	Fe T-4B2B3	1.07-012	2.089+000	1.000	1.000	2.22-012	
	Fe T-4A6A	9.95-013	2.089+000	1.000	1.000	2.08-012	
	Fe 3T-2C	8.80-013	2.089+000	1.000	1.000	1.84-012	
	Fe 5T-3E	8.30-013	2.089+000	1.000	1.000	1.73-012	
	Fe X-2A	7.77-013	2.089+000	1.000	1.000	1.62-012	
	Fe 3X-1A	5.35-013	2.089+000	1.000	1.000	1.12-012	
	Fe 7X-2A	1.73-013	2.089+000	1.000	1.000	3.61-013	

Ni58(N,P)Co58	Ni R-3C	3.96-013	1.751+000	1.000	1.000	6.93-013	12.
	Ni 3R-2C	6.38-013	1.751+000	1.000	1.000	1.12-012	
	Ni 5R-2A	9.45-013	1.751+000	1.000	1.000	1.65-012	
	Ni 7R-2A	8.97-013	1.751+000	1.000	1.000	1.57-012	
	Ni S-2C	1.05-012	1.751+000	1.000	1.000	1.84-012	
	Ni 3S-2C	1.01-012	1.751+000	1.000	1.000	1.78-012	
	Ni 5S-2B	1.06-012	1.751+000	1.000	1.000	1.86-012	
	Ni T-4B2B3	9.99-013	1.751+000	1.000	1.000	1.75-012	
	Ni T-4A6A	1.07-012	1.751+000	1.000	1.000	1.88-012	
	Ni 3T-2C	9.16-013	1.751+000	1.000	1.000	1.60-012	
	Ni 5T-3E	1.03-012	1.751+000	1.000	1.000	1.81-012	
	Ni X-2A	9.48-013	1.751+000	1.000	1.000	1.66-012	
	Ni 3X-1A	6.91-013	1.751+000	1.000	1.000	1.21-012	
	Ni 7X-2A	2.53-013	1.751+000	1.000	1.000	4.44-013	

* End-of-irradiation (EOI) specific activity.

** The counting statistics are estimated by Matsumoto to be in the range of + 3 to 5% (one sigma) for the Mn54, Co59, Zr95, and Sn125 and + 8 to 10% for the Co58. The uncertainties (EOI) in the measured content (weight) of Fe54, Zr94, Sn124, and Ni58 are estimated by Matsumoto to be + 6%, + 0.5%, + 3% and + 6%, respectively. An EOI value for Co59 was not determined; see Appendix B. The counting statistics and isotopic uncertainties were combined in quadrature to obtain the one sigma measured reaction rate uncertainties. Uncertainties associated with target atom and reaction product atom burn-in and burn-out correction factors are not included; see report text and Appendix C. The specified uncertainty applies for for all measured values for each set of reaction rates; e.g., all Zr94(n,g)Zr95 specimens Zr R-3C thru Zr 7X-2A.

APPENDIX B

N-REACTOR PRESSURE TUBE GAMMA SPECTROMETRY AND ELEMENTAL ANALYSIS RESULTS

APPENDIX B

N REACTOR PROCESS (PRESSURE) TUBE GAMMA SPECTROMETRY AND ELEMENTAL ANALYSIS RESULTS

Attached are gamma spectrometric and elemental (ICP) analysis results for specimens sampled from tubes 2954 and 3053 plus other information associated with these results. The attachments include:

1. Tabulations of the gamma spectrometrically measured specimens with their axial locations, target element compositions used, weight of each specimen, radionuclides measured, nominal gamma photopeak energies and half-lives used, results in units of disintegrations per second per mg (dps/mg) of target element corrected for radioactive decay to a reference date-time of January 7, 1987, 12:00 noon. Data for tubes 2954 and 3053 are listed in separate 2 page tables;
2. Tabulation of the zircaloy specimens analyzed by inductive coupled plasma (ICP) optical emission spectrometry for alloy and impurity element contents of interest, (with analogous original archive analytical data tabulated for comparison).
3. A tabulated summary of the activation reactions measured and the activation products and half-lives used.
4. The procedure used to clean (etch) the specimens in preparation for weighing and gamma spectrometry;

TABLE 1

GAMMA SPECTROMETRIC RESULTS: N-REACTOR ZIRCALOY-2 PRESSURE TUBE (2934) SECTIONS

Zircaloy-2 Pressure Tube Section "26-2A6" Ends vs. Center

SPECIMEN	LOC. IN.	WT. (MG)	NUCLIDE	T. COMP.	Peak (keV)	HALF-LIFE	DPS/MG (T)
ZR 2B-2B	132.8	48.8	MN-54	0.001370	835	312.70	6.784E+04
ZR 2B-2B	132.8	48.8	CO-58	0.000490	811	70.80	1.031E+06
ZR 2B-2B	132.8	48.8	ZR-95	0.982500	724	64.02	3.831E+05
ZR 2B-2B	132.8	48.8	CO-60	0.000114	1332	1925.18	9.476E+06
ZR 2B-2B	132.8	48.8	SB-125	0.015000	428	1011.71	9.692E+05
ZR F-2B	172.6	45.275	MN-54	0.001370	835	312.70	3.427E+05
ZR F-2B	172.6	45.275	CO-58	0.000490	811	70.80	4.749E+06
ZR F-2B	172.6	45.275	ZR-95	0.982500	724	64.02	1.627E+06
ZR F-2B	172.6	45.275	CO-60	0.000114	1332	1925.18	3.208E+07
ZR F-2B	172.6	45.275	SB-125	0.015000	428	1011.71	4.115E+06
ZR 3F-2B	203.5	34.57	MN-54	0.001370	835	312.70	4.992E+05
ZR 3F-2B	203.5	34.57	CO-58	0.000490	811	70.80	6.212E+06
ZR 3F-2B	203.5	34.57	ZR-95	0.982500	724	64.02	2.466E+06
ZR 3F-2B	203.5	34.57	CO-60	0.000114	1332	1925.18	4.860E+07
ZR 3F-2B	203.5	34.57	SB-125	0.015000	428	1011.71	5.946E+06
ZR 5F-2A1	229.8	30.585	MN-54	0.001370	835	312.70	6.060E+05
ZR 5F-2A1	229.8	30.585	CO-58	0.000490	811	70.80	7.251E+06
ZR 5F-2A1	229.8	30.585	ZR-95	0.982500	724	64.02	2.705E+06
ZR 5F-2A1	229.8	30.585	CO-60	0.000114	1332	1925.18	3.399E+07
ZR 5F-2A1	229.8	30.585	SB-125	0.015000	428	1011.71	6.651E+06
ZR 7F-2C	264.0	54.835	MN-54	0.001370	835	312.70	6.264E+05
ZR 7F-2C	264.0	54.835	CO-58	0.000490	811	70.80	6.504E+06
ZR 7F-2C	264.0	54.835	ZR-95	0.982500	724	64.02	2.911E+06
ZR 7F-2C	264.0	54.835	CO-60	0.000114	1332	1925.18	6.333E+07
ZR 7F-2C	264.0	54.835	SB-125	0.015000	428	1011.71	7.266E+06
ZR 26-2A6	299.0	48.895	MN-54	0.001370	835	312.70	6.769E+05
ZR 26-2A6	299.0	48.895	CO-58	0.000490	811	70.80	7.047E+06
ZR 26-2A6	299.0	48.895	ZR-95	0.982500	724	64.02	2.974E+06
ZR 26-2A6	299.0	48.895	CO-60	0.000114	1332	1925.18	6.478E+07
ZR 26-2A6	299.0	48.895	SB-125	0.015000	428	1011.71	7.271E+06
ZR 26-2A6e	299.0	16.360	MN-54	0.001370	835	312.70	6.546E+05 *
ZR 26-2A6e	299.0	16.360	CO-58	0.000490	811	70.80	6.222E+06 *
ZR 26-2A6e	299.0	16.360	ZR-95	0.982500	724	64.02	3.018E+06 *
ZR 26-2A6e	299.0	16.360	CO-60	0.000114	1332	1925.18	6.993E+07 *
ZR 26-2A6e	299.0	16.360	SB-125	0.015000	428	1011.71	7.294E+06 *
ZR 26-2A6c	299.0	32.520	MN-54	0.001370	835	312.70	6.529E+05 *
ZR 26-2A6c	299.0	32.520	CO-58	0.000490	811	70.80	6.600E+06 *
ZR 26-2A6c	299.0	32.520	ZR-95	0.982500	724	64.02	2.945E+06 *
ZR 26-2A6c	299.0	32.520	CO-60	0.000114	1332	1925.18	6.221E+07 *
ZR 26-2A6c	299.0	32.520	SB-125	0.015000	428	1011.71	7.234E+06 *

* The end (e) sections from the 26-2A6 section were cut off and combined for counting and the center section (c) counted separately. Results from these counts show that the dark material not removed from the ends by ultra-sonic acid etching does not does not contain a significant quantity of deposited activation products.

ZR 3G-3C	316.3	30.47	MN-54	0.001370	835	312.70	6.590E+05
ZR 3G-3C	316.3	30.47	CO-58	0.000490	811	70.80	6.670E+06
ZR 3G-3C	316.3	30.47	ZR-95	0.982500	724	64.02	2.922E+06
ZR 3G-3C	316.3	30.47	CO-60	0.000114	1332	1925.18	6.256E+07
ZR 3G-3C	316.3	30.47	SB-125	0.015000	428	1011.71	7.152E+06

TABLE 1 (Cont'd.)

SPECIMEN	LOC. IN.	WT. (MG)	NUCLIDE	T. COMP.	Peak (keV)	HALF-LIFE	DPS/MB (T)
ZR J-4D2	343.4	34.995	MN-54	0.001370	835	312.70	6.402E+03
ZR J-4D2	343.4	34.995	CO-58	0.000490	811	70.80	7.313E+06
ZR J-4D2	343.4	34.995	ZR-95	0.982500	724	64.02	2.606E+06
ZR J-4D2	343.4	34.995	CO-60	0.000114	1332	1925.18	5.118E+07
ZR J-4D2	343.4	34.995	SB-125	0.015000	428	1011.71	6.721E+06
ZR J-4B3A	360.2	45.86	MN-54	0.001370	835	312.70	6.142E+03
ZR J-4B3A	360.2	45.86	CO-58	0.000490	811	70.80	7.370E+06
ZR J-4B3A	360.2	45.86	ZR-95	0.982500	724	64.02	2.661E+06
ZR J-4B3A	360.2	45.86	CO-60	0.000114	1332	1925.18	5.203E+07
ZR J-4B3A	360.2	45.86	SB-125	0.015000	428	1011.71	6.764E+06
ZR J-2A5	369.0	21.195	MN-54	0.001370	835	312.70	5.578E+04 **
ZR J-2A5	369.0	21.195	CO-58	0.000490	811	70.80	8.096E+03 **
ZR J-2A5	369.0	21.195	ZR-95	0.982500	724	64.02	4.914E+01 **
ZR J-2A5	369.0	21.195	CO-60	0.000114	1332	1925.18	3.166E+07 **
ZR J-2A5	369.0	21.195	SB-125	0.015000	428	1011.71	3.464E+06 **
ZR J-2A6B1	369.0	24.85	MN-54	0.001370	835	312.70	5.999E+03
ZR J-2A6B1	369.0	24.85	CO-58	0.000490	811	70.80	7.134E+06
ZR J-2A6B1	369.0	24.85	ZR-95	0.982500	724	64.02	2.744E+06
ZR J-2A6B1	369.0	24.85	CO-60	0.000114	1332	1925.18	5.513E+07
ZR J-2A6B1	369.0	24.85	SB-125	0.015000	428	1011.71	6.840E+06
** Sample section J-2A5 is obviously a bad sample. Sections J-4B3A and J-2A6B1 taken at, or near the location for J-2A5 are consistent with expected results.							
ZR 3J-2D	395.3	42.54	MN-54	0.001370	835	312.70	5.827E+03
ZR 3J-2D	395.3	42.54	CO-58	0.000490	811	70.80	6.596E+06
ZR 3J-2D	395.3	42.54	ZR-95	0.982500	724	64.02	2.760E+06
ZR 3J-2D	395.3	42.54	CO-60	0.000114	1332	1925.18	5.808E+07
ZR 3J-2D	395.3	42.54	SB-125	0.015000	428	1011.71	6.801E+06
ZR 5J-4C	415.8	42.47	MN-54	0.001370	835	312.70	5.461E+03
ZR 5J-4C	415.8	42.47	CO-58	0.000490	811	70.80	6.201E+06
ZR 5J-4C	415.8	42.47	ZR-95	0.982500	724	64.02	2.628E+06
ZR 5J-4C	415.8	42.47	CO-60	0.000114	1332	1925.18	5.511E+07
ZR 5J-4C	415.8	42.47	SB-125	0.015000	428	1011.71	6.441E+06
ZR 5J-2C	433.8	41.355	MN-54	0.001370	835	312.70	5.082E+03
ZR 5J-2C	433.8	41.355	CO-58	0.000490	811	70.80	6.632E+06
ZR 5J-2C	433.8	41.355	ZR-95	0.982500	724	64.02	2.409E+06
ZR 5J-2C	433.8	41.355	CO-60	0.000114	1332	1925.18	5.042E+07
ZR 5J-2C	433.8	41.355	SB-125	0.015000	428	1011.71	6.085E+06
ZR 3K-2C	480.0	41.32	MN-54	0.001370	835	312.70	3.542E+03
ZR 3K-2C	480.0	41.32	CO-58	0.000490	811	70.80	4.725E+06
ZR 3K-2C	480.0	41.32	ZR-95	0.982500	724	64.02	1.714E+06
ZR 3K-2C	480.0	41.32	CO-60	0.000114	1332	1925.18	3.518E+07
ZR 3K-2C	480.0	41.32	SB-125	0.015000	428	1011.71	4.329E+06
ZR 6K-4B	515.1	46.155	MN-54	0.001370	835	312.70	2.176E+03
ZR 6K-4B	515.1	46.155	CO-58	0.000490	811	70.80	3.238E+06
ZR 6K-4B	515.1	46.155	ZR-95	0.982500	724	64.02	1.032E+06
ZR 6K-4B	515.1	46.155	CO-60	0.000114	1332	1925.18	2.114E+07
ZR 6K-4B	515.1	46.155	SB-125	0.015000	428	1011.71	2.789E+06
ZR 6K-2A	531.6	33.705	MN-54	0.001370	835	312.70	1.320E+03
ZR 6K-2A	531.6	33.705	CO-58	0.000490	811	70.80	1.807E+06
ZR 6K-2A	531.6	33.705	ZR-95	0.982500	724	64.02	6.486E+05
ZR 6K-2A	531.6	33.705	CO-60	0.000114	1332	1925.18	1.367E+07
ZR 6K-2A	531.6	33.705	SB-125	0.015000	428	1011.71	1.904E+06

TABLE 2

GAMMA SPECTROMETRIC RESULTS: N-REACTOR ZIRCALOY-2 PRESSURE TUBE (3053) SECTIONS

SPECIMEN	LOC. IN.	WT. (MG)	NUCLIDE	T. COMP.	Peak(keV)	HALF-LIFE	DPS/MG (T)
ZR R-3C	151.7	37.573	MN-54	0.001135	835	312.70	1.827E+05
ZR R-3C	151.7	37.573	CO-58	0.000500	811	70.80	2.774E+06
ZR R-3C	151.7	37.573	ZR-95	0.982500	724	64.02	8.250E+05
ZR R-3C	151.7	37.573	CO-60	0.000025	1332	1925.18	2.968E+08
ZR R-3C	151.7	37.573	SB-125	0.014800	428	1011.71	2.383E+06
ZR JR-2C	183.0	35.895	MN-54	0.001135	835	312.70	3.419E+05
ZR JR-2C	183.0	35.895	CO-58	0.000500	811	70.80	4.466E+06
ZR JR-2C	183.0	35.895	ZR-95	0.982500	724	64.02	1.850E+06
ZR JR-2C	183.0	35.895	CO-60	0.000025	1332	1925.18	5.680E+08
ZR JR-2C	183.0	35.895	SB-125	0.014800	428	1011.71	4.657E+06
ZR SR-2A	213.0	26.770	MN-54	0.001135	835	312.70	5.452E+05
ZR SR-2A	213.0	26.770	CO-58	0.000500	811	70.80	6.622E+06
ZR SR-2A	213.0	26.770	ZR-95	0.982500	724	64.02	2.523E+06
ZR SR-2A	213.0	26.770	CO-60	0.000025	1332	1925.18	6.968E+08
ZR SR-2A	213.0	26.770	SB-125	0.014800	428	1011.71	6.367E+06
ZR 7R-2A	231.6	37.860	MN-54	0.001135	835	312.70	6.000E+05
ZR 7R-2A	231.6	37.860	CO-58	0.000500	811	70.80	6.286E+06
ZR 7R-2A	231.6	37.860	ZR-95	0.982500	724	64.02	2.761E+06
ZR 7R-2A	231.6	37.860	CO-60	0.000025	1332	1925.18	7.364E+08
ZR 7R-2A	231.6	37.860	SB-125	0.014800	428	1011.71	6.722E+06
ZR S-2C	250.0	26.670	MN-54	0.001135	835	312.70	6.360E+05
ZR S-2C	250.0	26.670	CO-58	0.000500	811	70.80	7.359E+06
ZR S-2C	250.0	26.670	ZR-95	0.982500	724	64.02	2.900E+06
ZR S-2C	250.0	26.670	CO-60	0.000025	1332	1925.18	7.514E+08
ZR S-2C	250.0	26.670	SB-125	0.014800	428	1011.71	7.000E+06
ZR 3S-2C	283.9	27.800	MN-54	0.001135	835	312.70	6.509E+05
ZR 3S-2C	283.9	27.800	CO-58	0.000500	811	70.80	7.109E+06
ZR 3S-2C	283.9	27.800	ZR-95	0.982500	724	64.02	2.982E+06
ZR 3S-2C	283.9	27.800	CO-60	0.000025	1332	1925.18	7.718E+08
ZR 3S-2C	283.9	27.800	SB-125	0.014800	428	1011.71	7.349E+06
ZR 5S-2B	312.8	38.300	MN-54	0.001135	835	312.70	6.615E+05
ZR 5S-2B	312.8	38.300	CO-58	0.000500	811	70.80	7.447E+06
ZR 5S-2B	312.8	38.300	ZR-95	0.982500	724	64.02	2.957E+06
ZR 5S-2B	312.8	38.300	CO-60	0.000025	1332	1925.18	7.400E+08
ZR 5S-2B	312.8	38.300	SB-125	0.014800	428	1011.71	7.156E+06
ZR T-4B2B3	351.2	45.785	MN-54	0.001135	835	312.70	6.662E+05
ZR T-4B2B3	351.2	45.785	CO-58	0.000500	811	70.80	6.997E+06
ZR T-4B2B3	351.2	45.785	ZR-95	0.982500	724	64.02	2.573E+06
ZR T-4B2B3	351.2	45.785	CO-60	0.000025	1332	1925.18	6.519E+08
ZR T-4B2B3	351.2	45.785	SB-125	0.014800	428	1011.71	6.593E+06
ZR T-4A6A	360.7	28.995	MN-54	0.001135	835	312.70	6.222E+05
ZR T-4A6A	360.7	28.995	CO-58	0.000500	811	70.80	7.522E+06
ZR T-4A6A	360.7	28.995	ZR-95	0.982500	724	64.02	2.612E+06
ZR T-4A6A	360.7	28.995	CO-60	0.000025	1332	1925.18	6.685E+08
ZR T-4A6A	360.7	28.995	SB-125	0.014800	428	1011.71	6.681E+06

TABLE 2 (Cont'd.)

GAMMA SPECTROMETRIC RESULTS: N-REACTOR ZIRCALOY-2 PRESSURE TUBE (3053) SECTIONS

SPECIMEN	LOC. IN.	WT. (MG)	NUCLIDE	T. COMP.	Peak(keV)	HALF-LIFE	DPS/MG (T)
ZR 3T-2C	393.7	43.230	MN-54	0.001135	835	312.70	5.501E+05
ZR 3T-2C	393.7	43.230	CO-58	0.000500	811	70.80	6.417E+06
ZR 3T-2C	393.7	43.230	ZR-95	0.982500	724	64.02	2.762E+06
ZR 3T-2C	393.7	43.230	CO-60	0.000025	1332	1925.18	6.921E+08
ZR 3T-2C	393.7	43.230	SB-125	0.014800	428	1011.71	6.720E+06
ZR 5T-3E	412.4	51.840	MN-54	0.001135	835	312.70	5.194E+05
ZR 5T-3E	412.4	51.840	CO-58	0.000500	811	70.80	7.236E+06
ZR 5T-3E	412.4	51.840	ZR-95	0.982500	724	64.02	2.630E+06
ZR 5T-3E	412.4	51.840	CO-60	0.000025	1332	1925.18	6.620E+08
ZR 5T-3E	412.4	51.840	SB-125	0.014800	428	1011.71	6.395E+06
ZR X-2A	440.3	53.885	MN-54	0.001135	835	312.70	4.859E+05
ZR X-2A	440.3	53.885	CO-58	0.000500	811	70.80	6.642E+06
ZR X-2A	440.3	53.885	ZR-95	0.982500	724	64.02	2.249E+06
ZR X-2A	440.3	53.885	CO-60	0.000025	1332	1925.18	5.841E+08
ZR X-2A	440.3	53.885	SB-125	0.014800	428	1011.71	5.696E+06
ZR 3X-1A	479.2	35.205	MN-54	0.001135	835	312.70	3.345E+05
ZR 3X-1A	479.2	35.205	CO-58	0.000500	811	70.80	4.844E+06
ZR 3X-1A	479.2	35.205	ZR-95	0.982500	724	64.02	1.673E+06
ZR 3X-1A	479.2	35.205	CO-60	0.000025	1332	1925.18	4.549E+08
ZR 3X-1A	479.2	35.205	SB-125	0.014800	428	1011.71	4.272E+06
ZR 7X-2A	534.2	25.545	MN-54	0.001135	835	312.70	1.081E+05
ZR 7X-2A	534.2	25.545	CO-58	0.000500	811	70.80	1.775E+05
ZR 7X-2A	534.2	25.545	ZR-95	0.982500	724	64.02	5.306E+05
ZR 7X-2A	534.2	25.545	CO-60	0.000025	1332	1925.18	1.749E+08
ZR 7X-2A	534.2	25.545	SB-125	0.014800	428	1011.71	1.611E+06

TABLE 3

RESULTS FROM ANALYSES OF ZIRCALOY TUBE SECTIONS BY ICP OPTICAL EMISSION SPECTROMETRY

SAMPLE I.D.	Zr	Fe	Sn	Cr	Ni	Co*	Cu*
3053 B	98.81	.114	1.49	.099	.052	9	<0.1
3053 T	96.83	.113	1.47	.097	.049	51	<0.1
3053 Avg.	97.82	.1135	1.48	.098	.0505	30	<0.1
3053 Orig.	NG	.117	1.47	.107	.047	<20	<20
3053 Used**	98.25	.1135	1.48	N/U	.050	25	N/U
2954 B	94.70	.135	1.51	.092	.049	121	0.3
2954 T	96.22	.137	1.51	.093	.049	108	<0.1
2954 A3B	96.03	.137	1.48	.092	.049	114	<0.1
2954 5L-1A	95.61	.140	1.49	.091	.048	113	<0.1
2954 Avg.	95.64	.1372	1.498	.092	.0488	114	<0.1
2954 Orig.	NG	.132	1.47	.106	.048	<10	<20
2954 Used**	98.25	.137	1.50	N/U	.049	114	N/U

* Values for Co and Cu are in units of parts per million (ppm), values for the other elements are in weight percent.

** These values were used in the calculation of the dps/ag values listed in Table 1 and Table 2. For zirconium (Zr), the nominal value of 98.25% was used and has an estimated (one sigma) uncertainty of less than 0.5%. The measured analytical values tabulated which appear to be inconsistent with 98.25% value used, merely reflect expected limitations of the ICP analysis method which in this case was had a systematically low bias for analyzing the Zr. The ICP analyses were done in a manner to optimize the reliability for measuring the minor and trace elements using solutions that had extremely high concentrations of Zr in a range where the instrument is not expected to maintain a close linear response with the element concentration. Therefore, the tabulated measured values for Zr should be viewed as only qualitative/semi-quantitative results data.

Note: Although the elemental analysis results for the Co in the tube 2954 sample sections appear to be quite consistent, these values are not compatible with the gamma spectrometry results both on the basis of the Zr-95 and Sb-125 thermal neutron activation product results and previous flux measurements made in N Reactor with calibrated dosimeters. Using average thermal neutron flux values based on the Zr-95 and Sb-125 measured calculated values for the cobalt content of the tubes by "neutron activation analysis" are 0.78 ppm and 2.01ppm for tubes 2954 and 3053 respectively. The cause for the high results for cobalt in the ICP analyses is not understood at this time.

ATTACHMENT 4

TABLE 4

ACTIVATION REACTIONS MEASURED BY GAMMA SPECTROMETRY

<u>Reactions</u>	<u>Product Observed</u>	<u>Half-life (d)</u>
Fe-54(n,p)Mn-54	Mn-54	312.7 (5)*
Ni-58(n,p)Co-58	Co-58	70.80 (15)*
Zr-94(n,g)Zr-95	Zr-95	64.02 (30)*
Co-59(n,g)Co-60	Co-60	1925.18 (73)*
Sn-124(n,g)Sn-125 <u>b-</u> Sb-125	Sb-125	1011.71 (1461)*

* Numbers in parentheses represent the uncertainties in the last digits given in the values, e.g. 1011.71 (1461) is equivalent to 1011.71 +/-14.61.

ATTACHMENT 5

PROCEDURE FOR CLEANING AND WEIGHING SPECIMENS FOR GAMMA SPECTROMETRY

1. Each specimen had been transferred to a labeled clean 2 dram glass vial in the Shielded Analytical Laboratory (SAL), these were transferred to a radiochemistry laboratory for cleaning and weighing.
2. One ml of reagent grade acetone was added to each specimen containing vial and the vial agitated in an ultrasonic cleaner for 5 minutes followed by a second acetone rinse. Acetone was used to remove any oils that may have gotten on the specimens during the sectioning operations.
3. The specimen was then rinsed thoroughly with distilled water.
4. Approximately 1 ml of a 8M nitric acid - 1.2M hydrofluoric acid mixture was added to the specimen and the vial agitated in an ultrasonic cleaner for 10 minutes. Gas bubbles could be seen forming and escaping from the specimens surfaces as evidence of the chemical etching reaction. The ends of the specimens tended to remain dark even though the etching reaction was vigorous enough to completely dissolve sharp shards of metal left on some of the samples from the sectioning operation. In a few cases 12M hydrochloric - 3M hydrofluoric acid mixture was tried. Although the reaction was more vigorous in this mixture the ends of the specimens, representing the inner and outer surfaces of the tube, still remained dark. A test made on specimen 26-2A6 from tube 2954 by analyzing the whole specimen then shearing off both ends, combining the ends in a single sample for analysis and comparing these results with analysis of the center section showed no significant differences, (see Table 1).
5. The etched specimen was rinsed with distilled water followed by an acetone rinse and air dried.
6. The specimens were weighed on a Mettler Model AE 163 analytical balance using the 30 gram range with 5 place digital readout.
7. After weighing the samples were transferred to custom made polyethylene vials designed for holding dosimeter samples for gamma spectrometry in C & A's automated sample changer positioner (ASC-P) system.
8. The manual operation mode of the ASC-P was used in analyzing these samples due to a breakdown of the sample changer.

APPENDIX C

EFFECTS OF TRANSMUTATIONS ON N REACTOR PRESSURE TUBE DOSIMETRY

APPENDIX C

EFFECTIVE NEUTRON CROSS SECTIONS

Both neutron cross sections, $\sigma(E)$, and neutron flux intensity, $\phi(E)$, vary as a function of neutron energy; moreover, they often exhibit opposite dependencies on neutron energy, within a range of energies. Thus, the values of spectrum averaged neutron cross sections can be vastly different for different spectra. The spectrum averaged cross section, (a) σ , can be defined as follows:

$$\sigma = \frac{\int_0^{\infty} \sigma(E) \phi(E) dE}{\int_0^{\infty} \phi(E) dE}$$

For practical purposes, the upper and lower limits of integration can be replaced by energy limits, E_1 , and E_2 ; E_1 is often chosen as 10^{-4} eV, and E_2 as 18 MeV.

In order to reduce some of the large dependency of the quoted cross section on the shape of the neutron spectrum in which it was measured, we could define an effective cross section, σ_{eff} , as follows:

$$\sigma_{\text{eff}} = \frac{\int_0^{\infty} \sigma(E) \phi(E) dE}{\int_{E_1}^{E_2} \phi(E) dE} \quad (1)$$

Where the energy limits would be chosen such that $\int_{E_1}^{E_2} \sigma(E) \phi(E) dE$ is 80% (or 90%, or some other fraction) of $\int_0^{\infty} \sigma(E) \phi(E) dE$.¹ This approach, however, could become very complicated and confusing.

A simpler approach is possible because we can classify many nuclear reactions as either "low-energy" (thermal) or "high-energy" (fast) reactions. Thus, we can rewrite Equation 1 in the following two ways:

(a) Also referred to as $\sigma(\text{Total})$.

$$\sigma(E < E_u) = \int_0^{\infty} \sigma(E) \phi(E) dE / \int_0^{E_u} \phi(E) dE \quad (1a)$$

$$\sigma(E > E_l) = \int_0^{\infty} \sigma(E) \phi(E) dE / \int_{E_l}^{\infty} \phi(E) dE \quad (1b)$$

It is not claimed that, even with careful choice of E_u or E_l , the resultant "effective cross sections" would be invariant in all neutron spectra; however, this approach does reduce the potential error in choosing a value for a cross section when the shape of the neutron spectrum is not well-defined.

For the present application, I have chosen $E_u = 0.414$ eV for the (n, γ) activation reactions, and $E_l = 0.183$ MeV for the (n,p), (n, α), and other "high-energy" reactions. The choice of E_u and E_l is rather arbitrary; however: 1) Kellogg, McElroy, and Suski (1987) reported data, from irradiations in N Reactor, from which $\sigma(E < 0.414$ eV) can readily be derived for the $Fe^{58}(n,\gamma)Fe^{59}$ and $Co^{59}(n,\gamma)Co^{60}$ activation reactions, (a) and 2) Dahl(b) calculated $\sigma(E > 0.183$ MeV) for $Fe^{54}(n,p)Mn^{54}$, $Ni^{58}(n,p)Co^{58m}$, and $Ni^{58}(n,p)Co^{58}$, in a normal N Reactor lattice position. These cross sections could be converted to averages over other energy ranges; however, the conversion would involve additional assumptions, and possible sources of error.

Table 1 outlines the calculations made to obtain $\sigma(E < 0.414$ eV) from the data reported by Kellogg, McElroy, and Suski (1987).

Other low-energy neutron cross sections of interest are not available in terms of $\sigma(E < 0.414$ eV); therefore, another source of cross section data must be used.

(a) Data on the $Fe^{54}(n,p)Mn^{54}$ reaction were also reported; however, the high-energy end of the neutron spectrum probably is much different than that near the pressure tube.

(b) Unpublished data, noted in my files (circa 1965).

TABLE 1. Calculation of $\text{Fe}^{58}(n,\gamma)\text{Fe}^{59}$ and $\text{Co}^{59}(n,\gamma)\text{Co}^{60}$ Cross Sections

Reaction	Reported $\sigma(\text{Total}), \text{Barns}$	Ratio $\frac{\phi(\text{Total})}{\phi(E < 0.414 \text{ eV})}$	Calculated $\sigma(E < 0.414 \text{ eV}), \text{Barns}$
$\text{Fe}^{58}(n,\gamma)\text{Fe}^{59}$	0.5337	1.157	0.617
$\text{Co}^{59}(n,\gamma)\text{Co}^{60}$	17.68	1.157	20.5

It is customary to quote low-energy neutron flux intensities and cross sections in terms of "thermal" (2200 m/s) neutrons. In principal, a sample is exposed to a known flux intensity of mono-energetic (2200 m/s) neutrons, and the "thermal" neutron cross section, $\sigma(2200 \text{ m/s})$, is derived from the induced activity (number of transmutations) and fluence $\phi(2200 \text{ m/s}) \times \text{time}$. If the sample is then exposed to an unknown flux spectrum, the induced activity, exposure time, and $\sigma(2200 \text{ m/s})$ can be used to derive the "thermal neutron" flux intensity; that is, the equivalent 2200 m/s flux intensity that would yield the same activation rate.

Table 2 shows the relationship between $\sigma(2200 \text{ m/s})$ and $\sigma(E < 0.414 \text{ eV})$ derived from the data of Kellogg, McElroy, and Suski (1987).

TABLE 2. Calculated Ratio of $\sigma(2200 \text{ m/s})$ to $\sigma(E < 0.414 \text{ eV})$

Reaction	$\sigma(E < 0.414 \text{ eV}), \text{Barns}$	$\sigma(2200 \text{ m/s}), (1) \text{Barns}$	Ratio $\frac{\sigma(E < 0.414 \text{ eV})}{\sigma(2200 \text{ m/s})}$
$\text{Fe}^{58}(n,\gamma)\text{Fe}^{59}$	0.617	1.187	0.520
$\text{Co}^{59}(n,\gamma)\text{Co}^{60}$	20.5	37.2	<u>0.551</u>
			Average = 0.535

(1) Zijp and Baard 1979.

For want of better information, we will assume that the ratio, $\sigma(E < 0.414 \text{ eV})/\sigma(2200 \text{ m/s}) = 0.535$, is a constant for the low-energy reactions of interest, in all N Reactor neutron spectra. It should be noted, however, if the low-energy cross section contains an epithermal component, this ratio may not be a constant.

R. E. Dahl(a) calculated $\sigma(E > 0.183 \text{ MeV})$ for the three high-energy neutron reactions of primary interest. At a normal lattice position (between pressure tubes) in N Reactor, the cross sections $\sigma(E > 0.183 \text{ MeV})$ are 29.8 mb for $\text{Ni}^{58}(n,p)\text{Co}^{58}$, 13.0 mb for $\text{Ni}^{58}(n,p)\text{Co}^{58m}$, and 27.5 mb for $\text{Fe}^{54}(n,p)\text{Mn}^{54}$.

The cross section, $\sigma(E)$, for the $\text{Fe}^{54}(n,p)\text{Mn}^{54}$ reaction, like that of most other high-energy reactions of interest, is not appreciable below about 2 MeV; thus, the activation reaction is (primarily) a response to the (first-flight) fission neutrons. Zijp and Baard (1979) list the fission-spectrum averaged activation cross section, σ_f , as 78.46mb, almost three times as large as $\sigma(E > 0.183 \text{ MeV})$. In the pressure tube, there will be a larger portion of first-flight fission neutrons; thus, $\sigma(E > 0.183 \text{ MeV})$ should increase, and may approach σ_f . We shall assume that $\sigma(E > 0.183 \text{ MeV})$ is 2/3 of σ_f for the Ni^{58} and $\text{Fe}^{54}(n,p)$ reactions, and equals σ_f for all other high-energy reactions; the tabulations of Zijp and Baard will be used for σ_f .

ESTIMATION OF NEUTRON FLUX INTENSITIES

In support of other programs, R. W. Twitchell(b) ran several reactor physics calculations to obtain neutron flux intensities in the N Reactor lattice. None of these calculations yield all of the information which is needed for the present situation; however, the data from three of the calculations can be combined to provide a reliable estimate of neutron flux intensities for the energy groups of interest, namely for neutrons with energies less than 0.414 eV, $\phi(E < 0.414 \text{ eV})$, and neutrons with energies greater than 0.183 MeV, $\phi(E > 0.183 \text{ MeV})$.

(a) Unpublished data, noted in my files (circa 1965).

(b) Then with United Nuclear Industries; now with Advanced Nuclear Fuels Corporation, Richland, Washington.

To assist in other comparisons, we also note the flux intensities for neutrons with energies greater than 0.1 MeV, $\phi(E > 0.1 \text{ MeV})$, and neutrons with energies greater than 1.0 MeV, $\phi(E > 1.0 \text{ MeV})$.

The three calculations were made using the MCNP Monte-Carlo reactor-physics code (Briesmeister 1986); copies of the output were transferred to the Battelle VAX computer network, and are archived under the file names (HEAT2, NPHI2, and SIC1) used in the following discussion.

HEAT2 used a two-dimensional representation of a quarter-lattice unit with Mark IV fuel. The pressure tube was modeled in Cell 71, but the flux intensities were only output for 3 energy groups: $\phi(E < 4.0 \text{ eV})$, $\phi(E > 50 \text{ KeV})$, and $\phi(\text{Total})$. The output of HEAT2 was used as input for the NPHI2 calculation. The output flux intensities of NPHI2 are given in 75 energy groups above 167 eV for the cells containing graphite, but are not listed for the pressure tube. However, Cell 57 represents a very small segment of graphite immediately adjacent to the pressure tube; thus, relative flux intensities of high-energy neutrons should be nearly the same as in the pressure tube.

The output of the SIC1 calculation contains energy group structure for neutrons at near-thermal energies; thus, it was used to ratio the $\phi(E < 0.414 \text{ eV})$ to $\phi(E < 4.0 \text{ eV})$ from the HEAT2 calculation. The ratios and resultant flux intensities are listed in Table 3; the detailed calculations are given on Page 17 of Laboratory Record Book BNW-52185.

NEUTRON FLUENCE

Fuel loadings in Pressure Tubes 2954 and 3053 generated a total of 16399 and 16999 MWD, at averages of 3875 and 4035 kWd per operating day (OD), (a) respectively. These power generation numbers equate to 4232 or 4213 OD. Using the peak neutron fluxes, listed in Table 3, and 4232 OD, we obtain life-time fluences of:

$$\phi(E < 0.414 \text{ eV}) = 3.2 \times 10^{22}/\text{cm}^2, \text{ and}$$

$$\phi(E > 0.183 \text{ MeV}) = 2.0 \times 10^{22}/\text{cm}^2$$

(a) N Reactor Tube Loading History, courtesy of C. C. MacInnis and W. S. Slippert (WHC).

TABLE 3. Estimated Neutron Flux Intensities at 4000 MW (Total Reactor Power)

Energy Group	Average Flux Intensity						Peak ⁽¹⁾ Flux Intensity/cm ² •s
	HEAT2, Cell 71 Flux Intensity	Ratio	NPHI2, Cell 57 Flux Intensity	Ratio	SIC1, Seg. 3 Flux Intensity		
E < 0.414 eV	-	-	-	0.825	$6.25 \times 10^{13}/\text{cm}^2\cdot\text{s}$		8.75×10^{13}
E < 4.0 eV	$7.57 \times 10^{13}/\text{cm}^2\cdot\text{s}$	-	-	1.0	-		1.06×10^{14}
E > 50 KeV	$5.22 \times 10^{13}/\text{cm}^2\cdot\text{s}$	1.0	-	-	-		7.31×10^{13}
E > 0.10 MeV	-	0.862	$4.50 \times 10^{13}/\text{cm}^2\cdot\text{s}$	-	-		6.30×10^{13}
E > 0.183 MeV	-	0.757	$3.95 \times 10^{13}/\text{cm}^2\cdot\text{s}$	-	-		5.53×10^{13}
E > 1.0 MeV	-	0.322	$1.68 \times 10^{13}/\text{cm}^2\cdot\text{s}$	-	-		2.35×10^{13}
Total	$1.84 \times 10^{14}/\text{cm}^2\cdot\text{s}$	-	-	-	-		2.58×10^{14}

1) A peak-to-average ratio of 1.40 was assumed.

DEPLETION OF Fe^{54} , Co^{59} , AND Ni^{58}

The depletion (burn-out) of these isotopes can be described by a differential equation of the form:

$$-\frac{dN}{dt} = N(\sigma_u \phi_u + \Sigma \sigma_l \phi_l) = N(\sigma_u + \sigma')\phi_u \quad (2)$$

Where: N = The concentration of the isotope (Fe^{54} , Co^{59} , or Ni^{58}) at time, t ,

$$\sigma_u = \sigma(E < 0.414 \text{ eV}) = 0.535 \cdot \sigma(2200 \text{ m/s}),$$

$$\phi_u = \phi(E < 0.414 \text{ eV}),$$

$$\sigma_l = \sigma(E > 0.183 \text{ MeV}),$$

$$\phi_l = \phi(E > 0.183 \text{ MeV}), \text{ and}$$

$$\sigma' = \Sigma \sigma_l \cdot \phi_l / \phi_u.$$

The solution of Equation 2 can be written as:

$$N = \langle N \rangle e^{-(\sigma_u + \sigma')\phi_u t} = \langle N \rangle e^{-(\sigma_u + \sigma')\Phi_u} \quad (3)$$

Where: $\langle N \rangle$ = The isotopic concentration at $t=0$,

$\Phi_u = \phi_u t = \phi(E < 0.414 \text{ eV})$, and the other quantities are defined, above.

All high-energy reactions, except $Fe^{54}(n,p)Mn^{54}$, have cross sections which are too small to contribute to the burnout of these isotopes. Zijp and Baard (1979) list σ_f for the $Fe^{54}(n,p)Mn^{54}$ reaction as 78.46 mb; the equivalent σ' (Equation 3) is 0.03 barns. Table 4 outlines the depletion calculations.

Normally, neutron flux spectral distributions are determined from the activation of flux monitor foils irradiated for only a short period of time; under these circumstances, depletion of the target isotope is of no importance. The TIMWH computer program is satisfactory, when used for short-time irradiations; however, improvements need to be incorporated to account for burnout of target and product isotopes.

TABLE 4. Depletion of Fe⁵⁴, Co⁵⁹, and Ni⁵⁸

Isotope	(n,γ) Reaction		N/<N>(2)
	$\sigma(2200 \text{ m/s})$ (1)	$\sigma(E < 0.414 \text{ eV})$	
Fe ⁵⁴	2.8	1.50	0.952
Co ⁵⁹	19.9 + 17	19.7	0.532
Ni ⁵⁸	4.4	2.35	0.928

(1) Handbook of Chemistry and Physics, 65th Edition, 1984-1985.

(2) $\phi(E < 0.414 \text{ eV}) = 3.2 \times 10^{22}/\text{cm}^2$.

Other transmutation paths, which may be important in some cases, are considered, below.

Ni⁵⁸ CONTRIBUTION TO Co⁶⁰ ACTIVITY

The (n,p) activation reaction of Ni⁵⁸ produces Co^{58m} and Co⁵⁸, both isotopes have large (n,γ) cross sections, yielding Co⁵⁹. The resultant Co⁵⁹ can then be transmuted to Co⁶⁰.

Zijp and Baard (1979) list the combined (n,p) cross section as 102.8 mb, averaged over the U²³⁵ fission spectrum. Assuming, as previously noted, that $\sigma(E > 0.183 \text{ MeV}) = 2/3$ of σ_f for the neutron spectrum in a pressure tube, $\sigma(E > 0.183 \text{ MeV}) = 68.5 \text{ mb}$; this value seems reasonable (though, perhaps, an upper limit) in view of Dahl's calculated cross section (29.8 + 13.0 = 42.8 mb) at a position between pressure tubes.

The amount of Ni⁵⁸ transmuted, ΔNi^{58} , to Co⁵⁸ can then be written as

$$\Delta\text{Ni}^{58} = [\text{Ni}^{58}] (1 - e^{-\sigma_1 \phi_1}) = 0.47 \text{ ppm}$$

$$\begin{aligned} \text{Where } [\text{Ni}^{58}] &= \text{The concentration of Ni}^{58} \\ &= 500 \text{ ppm} \times 68.27\% = 340 \text{ ppm} \end{aligned}$$

During an irradiation period, Co^{58} is lost due to both decay and transmutations to Co^{59} . The fraction ΔCo^{59} , which is transmuted to Co^{59} can be calculated as follows:

$$\Delta\text{Co}^{59} = \Delta\text{Ni}^{58} \frac{\sigma_u \phi_u}{\lambda + \sigma_u \phi_u} = 0.20 \text{ ppm}$$

Where σ_u = The transmutation cross section ($0.535 \times 1700 \text{ b} = 910 \text{ b}$)

$\phi_u = \phi(E < 0.414 \text{ eV})$, and

λ = The decay constant for Co^{58} ($1.12 \times 10^{-7}/\text{s}$).

The increase in Co^{59} concentration is, thus, less than 1% of the original Co^{59} impurity in Tube 3053.

THE CONTRIBUTION OF IRON TO Co^{60} ACTIVITY

Table 5 lists the natural abundances and cross sections of the three naturally occurring isotopes which form the transmutation chain:

TABLE 5. Data on Fe^{56} , Fe^{57} , and Fe^{58}

<u>Isotope</u>	<u>Abundance</u>	<u>$\sigma(2200 \text{ m/s}), \text{ b}$</u>	<u>$\sigma(E < 0.414 \text{ eV}), \text{ b}$</u>
Fe^{56}	0.918	2.5	1.34
Fe^{57}	0.021	2.5	1.34
Fe^{58}	0.003	1.23	0.66

Tube 2954 originally contained 1370 ppm of iron; of this 1258 ppm was Fe^{56} ; after $3.2 \times 10^{22}/\text{cm}^2$ ($E < 0.414 \text{ eV}$), 1205 ppm remains. Thus 53 ppm had been transmuted to Fe^{57} . To determine the Fe^{57} transmuted to Fe^{58} , we add one-half of the 53 ppm to the original 28.8 ppm.

$$55.3 \text{ ppm} \cdot \exp(-1.34 \times 10^{-24} \cdot 3.2 \times 10^{22}) = 53.0 \text{ ppm}$$

Thus, the concentration of Fe^{58} was increased from 4.1 ppm by $(55.3-53.0) = 2.3$ ppm. To this must be added the 0.27 ppm of Fe^{58} from decay of Co^{58} . The appropriate amount of Fe^{58} during the irradiation was then 5.3 ppm, and the amount transmuted to Fe^{59} is:

$$5.3 \text{ ppm } (1 - e^{-0.66 \times 10^{24} \cdot 3.2 \times 10^{22}}) = 0.11 \text{ ppm}$$

Even if all of the Fe^{59} decayed to Co^{59} , it wouldn't affect the Co^{60} activity.

CONTRIBUTIONS OF Ni^{60} AND Cu^{63} TO Co^{60} ACTIVITY

Tube 3053 contained 500 ppm of nickel (131 ppm of Ni^{60}). Zijp and Baard (1979) list σ_f as 2.44 mb for the $\text{Ni}^{60}(n,p)\text{Co}^{60}$ reaction; thus, the transmutation of Ni^{60} to Co^{60} would be about:

$$131 \text{ ppm } (1 - e^{-2.44 \times 10^{-27} \cdot 2.0 \times 10^{22}}) = 0.006 \text{ ppm}$$

Assuming a copper content of 20 ppm (14 ppm Cu^{63} , and $\sigma_f = 0.35$ mb (Zijp and Baard, 1979), the $\text{Cu}^{63}(n,\alpha)\text{Co}^{60}$ reaction would increase the Co^{60} constant by:

$$14 \text{ ppm } (1 - e^{-0.35 \times 10^{-27} \cdot 2.0 \times 10^{22}}) = 0.0001 \text{ ppm}$$

Thus, neither of these reactions would make a significant contribution to the Co^{60} activity.

REFERENCES

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Zijp, W. L., and J. H. Baard. 1979. Nuclear Data Guide For Reactor Neutron Metrology. EUR 7164 EN.

APPENDIX D

SPECTRAL ANALYSIS INFORMATION FOR PRESSURE TUBES 2954 AND 3053

APPENDIX D

SPECTRAL ANALYSIS INFORMATION FOR PRESSURE TUBES 2954 AND 3053

The information provided in this appendix was necessary for the flux-fluence and neutron spectrum analysis from the Pressure Tubes 2954 and 3053 samples. The data were provided by D. J. Trimble in Westinghouse Hanford Company Letter 8759140, dated June 30, 1987.

SPECTRAL SPECIMEN LIST

<u>Tube 2954</u>		<u>Tube 3053</u>	
<u>Specimen</u>	<u>Location, in.</u>	<u>Specimen</u>	<u>Location, in.</u>
1. 2B-2B	132.8	R-3C	151.7
2. F-2B	172.6	3R-2C	183.0
3. 3F-2B	203.5	5R-2A	213.0
4. 5F-2A	229.8	7R-2A	231.6
5. 7F-2C	264.0	S-2C	250.0
6. 2G-2A6	299.0	3S-2C	283.9
7. 3G-3C	316.3	5S-2B	312.8
8. J-4D2	343.4	T-4B2B3	351.2
9. J-2A5	369.0	T-4A6A	360.7
10. 3J-2D	395.3	3T-2C	393.7
11. 5J-4C	415.8	5T-3E	412.4
12. 5J-2C	433.8	X-2C	440.3
13. 3K-2C	480.0	3X-1A	479.2
14. 6K-4B	515.1	7X-2A	534.2
15. 6K-2A	531.6		

Tube No.	HT-285	Expt No.	H7C-1308-2- 3V-S25 Sp. Blend WC-8-58-1h	Billet No.		Extrusion No.		
Vendor Identification Number	H7C-130A-2	% of Cold Work	30	Weight Gain Mg/Dm ² -Max. 3h Mg.	18			
<p>Tube 2954</p> <p>N. P. R. ZIRCALOY TUBES</p> <p>CHEMICAL & CORROSION ANALYSIS SHEET</p>								
	%				PPM			
Chemical	Sn	Pb	Cr	Mn	N ₂	H ₂	Si	Co
Analysis	1.47	.132	.106	.048	42	12	49	*20
	PPM							
Chemical	W	C	O ₂	Al	Co	Pb	Hg	Mn
Analysis	*20	137	980	40	*10	*20	*10	*25
	PPM							
Chemical	Mo	Ti	V	Hf	B	Cd	U	Na
Analysis	*10	*20	*10	*.25	*.3	*.3	1.0	*10

Tube No.	HT-10	Mat. No.	H2C-1211	Spec. Blank Billet No.	wt. 5.51- 5.13	Extraction No.		
Vendor Identification Number	H2C-1211-3	% of Cold Work	16 %	Weight Gain Kg/Dm ² - Max. 36 Kg.	OK			
<p>Tube 3053</p> <p>N. P. R. ZIRCALOT TUBES</p> <p>CHEMICAL & CORROSION ANALYSIS SHEET</p>								
	S				FTM			
Chemical	Sn	Pb	Cr	Mn	N ₂	H ₂	Si	Cu
Analysis	1.47	.117	.107	.047	2.8	16	42	<20
	FTM							
Chemical	W	C	O ₂	Al	Co	Pb	Kg	Mn
Analysis	<20	110	980	58	<20	<20	<10	<25
	FTM							
Chemical	Mo	Ti	V	Hf	B	Cd	O	N ₂
Analysis	<10	<20	<10	<125	<.3	<.3	.3	<10

69 Group WIMS Calculated
Process Tube Integrated Fluxes

#. UPPER ENERGY(EV), LOWER ENERGY(EV), GROUP FLUX

1	1E+07	6.0655E+06	2.16291
2	6.0655E+06	3.679E+06	7.41348
3	3.679E+06	2.231E+06	16.1633
4	2.231E+06	1.353E+06	21.8726
5	1.353E+06	621000	23.2794
6	621000	500000	28.1035
7	500000	302500	21.4727
8	302500	183000	20.9976
9	183000	111000	17.9541
10	111000	67340	15.7063
11	67340	40850	14.6989
12	40850	24780	14.0255
13	24780	15030	13.7906
14	15030	9118	12.7249
15	9118	5530	12.2652
16	5530	3519.1	10.7969
17	3519.1	2239.45	10.6964
18	2239.45	1425.1	10.5707
19	1425.1	906.698	10.5611
20	906.698	367.262	20.6833
21	367.262	146.728	19.6109
22	146.728	75.5014	14.7824
23	75.5014	48.052	9.7166
24	48.052	27.7	11.6725
25	27.7	15.968	11.1976
26	15.968	9.877	9.62941
27	9.877	4	17.0275
28	4	3.3	3.65563
29	3.3	2.6	4.63816
30	2.6	2.1	4.2153
31	2.1	1.5	6.73741
32	1.5	1.3	2.92109
33	1.3	1.15	2.5156
34	1.15	1.123	.488669
35	1.123	1.097	.482952
36	1.097	1.071	.496394
37	1.071	1.045	.510882
38	1.045	1.02	.505744
39	1.02	.996	.499793
40	.996	.972	.514573
41	.972	.95	.485404
42	.95	.91	.918182
43	.91	.85	1.47149
44	.85	.78	1.88852
45	.78	.625	5.12695
46	.625	.5	5.84848
47	.5	.4	7.23573
48	.4	.35	5.42717
49	.35	.32	4.26478

PROCESS TUBE INTEGRATED FLUXES (CONT)

#, UPPER ENERGY(EV), LOWER ENERGY(EV), GROUP FLUX

50	.32	.3	3.46124
51	.3	.28	4.00513
52	.26	.25	7.62343
53	.25	.22	9.98617
54	.22	.18	18.4137
55	.18	.14	26.3637
56	.14	.1	36.6977
57	.1	.06	22.6636
58	.06	.067	15.8128
59	.067	.056	11.1885
60	.056	.05	9.91565
61	.05	.042	9.68126
62	.042	.035	8.07623
63	.035	.03	5.41517
64	.03	.025	5.00993
65	.025	.02	4.49078
66	.02	.015	3.83646
67	.015	.01	3.01829
68	.01	.005	2.00056
69	.005	0	.733573

TOTAL FLUX INTEGRAL = 669.37

APPENDIX E

WHC SUPPLIED BCL PRESSURE TUBE 1165 DATA

APPENDIX E

BCL DATA

The Battelle Columbus data used by the PNL-NDC for calculation and evaluation of flux-fluence for Pressure Tube 1165 were provided by WHC in three separate transmittals [References 1, 5, and 6]. To facilitate reference to the BCL data, the various data used in the present calculations have been extracted from these three referenced documents and are provided in this appendix.

Table 1 contains the radiometric (RM) counting data, reaction specific activities and reference date of analysis for the Pressure Tube 1165 samples.(a) These BCL data, after correction for composition and decay to end-of-irradiation (EOI), were used as input to the TIMWH Code to calculate the individual sample reaction rates (Appendix F). Table 2 provides the sample identification numbers, the sample locations, the iron impurity content, and the BCL [Reference 5] flux-fluence values. The fourteen Table 2 iron impurity values were used for the present PNL-NDC evaluation. Table 3 provides, for two samples, the BCL estimates of fast flux ($E > 1$ MeV) for the Pressure Tube 1165 Sample 777 location. The BCL best estimates of flux and fluence values for the Sample 777 location are given in Table 4.

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- (a) Those data required for the Pressure Tube 1165 re-evaluation studies (Appendices F and G) were excerpted from BCL counting log book results provided to D. J. Trimble (WHC) by M. P. Failey (BCL).
- [1] Letter, D. J. Trimble to T. Hikido. "Westinghouse Hanford Workslope Statement," Letter No. 8850380. January 18, 1988; Letter, D. J. Trimble to W. N. McElroy. "Fluence Calculations for Tube 1165," Letter No. 8850379. January 18, 1988.
- [5] Marshall and Scott 1984.
- [6] Lowry et al. 1985.

TABLE 1--BCL Sample Counting Data and Sample Specific Activities
for Pressure Tube 1165

Sample Identification	Nuclide ^a	⁵⁴ Mn, ⁶⁰ Co Specific Activity Measurements					
		ΔT (day)	Peak Area (gross counts) ^b	Counts per sec, T ₀ ^c	Wt (mg)	dps/mg	dps/mg averaged ^d
X	⁵⁴ Mn	466	717 (14)	3.36 x 10 ⁰	13.9	1.68 x 10 ²	1.32 x 10 ⁴
	⁶⁰ Co		97536 (<1) 86747 (<1)	1.92 x 10 ² 1.71 x 10 ²		1.31 x 10 ⁴ 1.32 x 10 ⁴	
2	⁵⁴ Mn	461	848 (7.0)	0.655	278.6	3.20 x 10 ⁻¹	5.83 x 10 ⁰
	⁶⁰ Co		25894 (<1) 23149 (<1)	8.49 x 10 ⁰ 7.59 x 10 ⁰		5.84 x 10 ⁰ 5.81 x 10 ⁰	
3	⁵⁴ Mn	461	689 (7.8)	0.533	152.4	4.76 x 10 ⁻¹	7.45 x 10 ⁰
	⁶⁰ Co		18273 (<1) 16075 (<1)	5.99 x 10 ⁰ 5.27 x 10 ⁰		7.52 x 10 ⁰ 7.37 x 10 ⁰	
4	⁵⁴ Mn	464	484 (15)	2.26 x 10 ⁰	5.2	3.02 x 10 ²	1.32 x 10 ⁴
	⁶⁰ Co		36273 (<1) 32673 (<1)	7.15 x 10 ¹ 6.44 x 10 ¹		1.31 x 10 ⁴ 1.33 x 10 ⁴	
88 short	⁵⁴ Mn	464	1120 (8.2)	5.22 x 10 ⁰	7.9	4.60 x 10 ²	8.56 x 10 ³
	⁶⁰ Co		35828 (<1) 32213 (<1)	7.06 x 10 ¹ 6.35 x 10 ¹		8.51 x 10 ³ 8.60 x 10 ³	
88 long	⁵⁴ Mn	464	1196 (7.1)	5.58 x 10 ⁰	8.3	4.67 x 10 ²	7.73 x 10 ³
	⁶⁰ Co		34051 (<1) 30509 (<1)	6.71 x 10 ¹ 6.01 x 10 ¹		7.70 x 10 ³ 7.75 x 10 ³	

TABLE 1--(Cont'd)

Sample Identification	Nuclide ^a	⁵⁴ Mn, ⁶⁰ Co Specific Activity Measurements					
		ΔT (day)	Peak Area (gross counts) ^b	Counts per sec, T_0 ^c	Wt (mg)	dps/mg	dps/mg averaged
6	⁵⁴ Mn	465	782 (11)	3.66×10^0	5.9	4.31×10^2	
	⁶⁰ Co		57647 (<1) 50954 (<1)	1.13×10^2 1.00×10^2		1.83×10^4 1.82×10^4	1.83×10^4
7	⁵⁴ Mn	465	1401 (5.7)	6.55×10^0	8.4	5.42×10^2	
	⁶⁰ Co		35302 (<1) 32354 (<1)	6.96×10^1 6.38×10^1		7.88×10^3 8.13×10^3	8.01×10^3
333	⁵⁴ Mn	465	391 (16)	3.05×10^{-2}	53.5	7.75×10^{-1}	
	⁶⁰ Co		32919 (<1) 29645 (<1)	1.08×10^1 9.74×10^0		3.87×10^1 3.88×10^1	3.88×10^1
AA	⁵⁴ Mn	465	981 (6.9)	4.59×10^0	5.5	5.79×10^2	
	⁶⁰ Co		20435 (<1) 18385 (<1)	4.03×10^1 3.62×10^1		6.73×10^3 7.05×10^3	6.89×10^3
44	⁵⁴ Mn	465	≤ 133	6.22×10^{-1}	5.0	$<1.69 \times 10^1$	
	⁶⁰ Co		52709 (<1) 46901 (<1)	1.04×10^2 9.24×10^1		3.98×10^3 3.94×10^3	3.96×10^3
777	⁵⁴ Mn	465	998 (6.8)	4.67×10^0	5.7	5.69×10^2	
	⁶⁰ Co		23092 (<1) 20912 (<1)	4.55×10^1 4.12×10^1		7.60×10^3 7.74×10^3	7.67×10^3
	¹²⁵ Sb		179720 (<1) 77006 (<1) 46664 (<1)	4.15×10^2 1.78×10^2 1.08×10^2		9.11×10^4 8.66×10^4 9.04×10^4	8.94×10^4
	⁹⁵ Zr	465	27211 34571	6.96×10^3 8.84×10^3	5.7	1.65×10^6 1.78×10^6	1.72×10^6

TABLE 1--(Cont'd)

Sample Identification	Nuclide ^a	⁵⁴ Mn, ⁶⁰ Co Specific Activity Measurements					
		T (day)	Peak Area (gross counts) ^b	Counts per sec, T ₀ ^c	Wt (mg)	dps/mg	dps/mg averaged ^d
JJ	⁵⁴ Mn	465	751 (8.8)	3.51×10^0	6.9	3.54×10^2	
	⁶⁰ Co		25428 (<1) 22803 (<1)	5.01×10^1 4.49×10^1		6.92×10^3 6.97×10^3	6.95×10^3
1	⁵⁴ Mn	464	1101 (5.6)	5.71×10^{-2}	318.1	2.44×10^{-2}	
	⁶⁰ Co		21658 (<1) 19471 (<1)	4.60×10^{-1} 4.13×10^{-1}		2.77×10^{-1} 2.77×10^{-1}	2.77×10^{-1}
1-first batch	⁵⁴ Mn	465	5284 (1.5)	2.75×10^{-2}	258.1	1.45×10^{-1}	
	⁶⁰ Co		1649 (4) 1420 (5)	2.22×10^{-2} 1.82×10^{-2}		1.64×10^{-2} 1.50×10^{-2}	1.57×10^{-2}
3-first batch	⁵⁴ Mn	466	97 (28)	2.12×10^{-3}	240.4	$<1.20 \times 10^{-3}$	
	⁶⁰ Co		630 622	$<1.27 \times 10^{-3}$ $<1.27 \times 10^{-3}$		$<1.01 \times 10^{-3}$ $<1.13 \times 10^{-3}$	

^aThe following nuclear parameters were used in the analysis of the samples.

Nuclide	Energy keV	Branching Intensity
⁵⁴ Mn	834	0.999
⁶⁰ Co	1173	0.999
⁶⁰ Co	1333	1.00
¹²⁵ Sb	428	0.296
¹²⁵ Sb	601	0.184
¹²⁵ Sb	636	0.112
⁹⁵ Zr	724	0.442
⁹⁵ Zr	757	0.548

^bNumber in parentheses is counting statistic in percent.

^cBackground Subtracted counts decayed to June 19, 1982.

^dAveraged value for nuclide with multiple counts.

TABLE 2
INITIAL DATA SUPPLIED BY WHC ON DCL
REPORTED RESULTS FOR PRESSURE TUBE 1165 SAMPLES
(Taken from Reference 5)

TABLE 6. FLUX AND FLUENCE ($E > 1$ MEV) RESULTS FOR SAMPLES
FROM THE N-REACTOR PRESSURE TUBE 1165

Specimen No.	Specimen Location(a)(ft)	Flux (n/cm ² /sec)	Fluence (n/cm ²)	Iron Content (weight %)
1	0	0	0	-
1	3.2	4.97×10^9	1.30×10^{18}	0.120
2	4.3	1.06×10^{10}	2.76×10^{18}	0.120
3	7.0	1.58×10^{10}	4.12×10^{18}	0.120
333	8.7	2.54×10^{10}	6.63×10^{18}	0.120
44	9.7	5.58×10^{11}	1.45×10^{20}	0.120
4	14.3	1.01×10^{13}	2.63×10^{21}	0.120
6	18.0	1.44×10^{13}	3.75×10^{21}	0.120
7	23.5	1.72×10^{13}	4.48×10^{21}	0.126
777	27.1	1.80×10^{13}	4.69×10^{21}	0.126
88A	32.8	1.45×10^{13}	3.78×10^{21}	0.126
88B	32.8	1.48×10^{13}	3.86×10^{21}	0.126
AA	36.5	1.83×10^{13}	4.77×10^{21}	0.126
JJ	40.9	1.20×10^{13}	3.13×10^{21}	0.117
X	45.8	5.74×10^{12}	1.50×10^{21}	0.117
111	53	0	0	-

Full Power = 3650 MW_t

Total MWD = 11,606,300 MW_t Days

Total operating seconds = 2.6046×10^8 sec.

(a) Location was measured from the pressure tube inlet.

BATTELLE - COLUMBUS

[Ref. 5] Marshall and Scott 1984.

TABLE 3
FINAL BCL REPORTED DATA FOR TWO SAMPLES
TAKEN FROM SAMPLE 777-PRESSURE TUBE 1165
(Taken from Reference 6)

TABLE 12. BENCHMARK RESULTS FOR TUBE 1165

Nuclide	Effective Cross Section (barns)	Activity (dps/mg) at Shutdown	Fast Flux ($E > 1$ Mev) at Full Power (n/cm ² sec)
Sample 1 Sb-125	2.328	$9.325 \times 10^4 \pm \sim 3\%$	2.673×10^{11}
Sample 1 Mn-54	0.123	$6.481 \times 10^2 \pm \sim 8.8\%$	1.682×10^{13}
Sample 2 Sb-125	2.328	$9.604 \times 10^4 \pm \sim 3\%$	2.753×10^{11}
Sample 2 Mn-54	0.123	$5.602 \times 10^2 \pm \sim 8.8\%$	1.454×10^{13}

[Ref. 6] Lowry et al. 1985.

TABLE 4
BCL BEST ESTIMATE OF FLUX-FLUENCE > 1 MEV
FOR PRESSURE TUBE 1165
(Taken from Reference 6)

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TABLE 13. SUMMARY OF BEST ESTIMATE FAST FLUXES ($E > 1$ MeV)
FOR TUBES 1165 AND 2566

Tube/ Reaction	Fast Flux ($E > 1$ MeV) at Full Power (n/cm^2 sec)	Fast Fluence (n/cm^2)
1165 Fe-54 (n,p) Mn-54	1.556×10^{13}	4.036×10^{21}
2566 Sn-124 (n, γ) Sb-125	1.637×10^{13}	3.228×10^{21}

[Ref. 6] Lowry et al. 1985.

APPENDIX F

TIMWH INPUT DATA AND DERIVED REACTION RATES
FOR N-REACTOR PRESSURE TUBE 1165 DOSIMETRY

APPENDIX F

TIMWH INPUT DATA AND DERIVED REACTION RATES FOR N-REACTOR PRESSURE TUBE 1165 DOSIMETRY

DATE AND TIME IN - 02/14/64 AT 12:00
DATE AND TIME OUT - 04/22/82 AT 12:00

EXPOSURE- MW SEC. MW HOURS MW DAYS
1.1034+009 3.0651+005 1.2771+004

TOTAL TIME IN REACTOR = 159408.0000 HRS
NORMALIZED TO POWER OF 4.07 MW

POWER (MW) WAS-

2.8500	.0000	2.8500	.0000	2.8500
.0000	2.8500	.0000	3.8050	.0000
3.8050	.0000	3.8050	.0000	3.8050
.0000	3.8050	.0000	3.8050	.0000
4.0890	.0000	4.0890	.0000	4.0890
.0000	4.0890	.0000	3.3480	.0000
3.3480	.0000	3.3480	.0000	3.9160
.0000	3.9160	.0000	3.9160	.0000
3.9160	.0000	3.9950	.0000	3.9950
.0000	3.9950	.0000	3.9950	.0000
4.0870	.0000	4.0870	.0000	4.0870
.0000	4.0870	.0000	3.4510	.0000
3.4510	.0000	3.4510	.0000	4.0210
.0000	4.0210	.0000	4.0210	.0000
3.9250	.0000	3.9250	.0000	3.9250
.0000	3.9250	.0000	3.3890	.0000
3.3890	.0000	3.3890	.0000	3.3890
.0000	3.3890	.0000	3.9000	.0000
3.9000	.0000	3.9000	.0000	3.9000
.0000	3.9000	.0000	4.0080	.0000
4.0080	.0000	4.0080	.0000	4.0470
.0000	4.0470	.0000	4.0470	.0000
3.8350	.0000	3.8350	.0000	3.8350
.0000	3.8350	.0000	3.8350	.0000
4.0950	.0000	4.0950	.0000	4.0950
.0000	4.0950	.0000	4.0950	

TIME (HRS) WAS-

720.0000	2952.0000	1682.0000	1992.0000	3000.0000
2867.0000	947.0000	689.0000	871.0000	696.0000
1784.0000	1320.0000	1152.0000	1190.0000	818.0000
370.0000	1479.0000	363.0000	380.0000	210.0000
630.0000	488.0000	1216.0000	452.0000	628.0000
181.0000	275.0000	1650.0000	756.0000	1969.0000
1073.0000	420.0000	948.0000	123.0000	555.0000

282.0000	912.0000	1149.0000	2904.0000	712.0000
419.0000	613.0000	875.0000	245.0000	883.0000
1483.0000	2521.0000	1305.0000	715.0000	451.0000
1037.0000	410.0000	550.0000	2433.0000	1755.0000
2086.0000	374.0000	4550.0000	1522.0000	2299.0000
1248.0000	2496.0000	1805.0000	185.0000	1207.0000
491.0000	2533.0000	270.0000	1434.0000	2215.0000
1289.0000	356.0000	1492.0000	192.0000	1779.0000
701.0000	1000.0000	1752.0000	1955.0000	389.0000
800.0000	435.0000	2589.0000	2040.0000	925.0000
1067.0000	1008.0000	397.0000	1451.0000	216.0000
1693.0000	2208.0000	1220.0000	2016.0000	1226.0000
1307.0000	866.0000	1871.0000	2577.0000	1208.0000
1640.0000	350.0000	1666.0000	2141.0000	1873.0000
1056.0000	1650.0000	518.0000	1258.0000	435.0000
552.0000	1672.0000	2036.0000	1554.0000	1023.0000
1015.0000	728.0000	7224.0000	1353.0000	2132.0000
925.0000	720.0000	797.0000	1134.0000	940.0000
232.0000	632.0000	227.0000	685.0000	

I	REACTION NAME	DECAY CONST. (Sec-1)	FISSION YLD FRACTION	SATURATION FACTOR	CORRECTION TO EOI	X FACTOR(1.0E-19)* (mg/atom)
1	FE54(N,P)MM54	2.567-008	1.000+000	2.243+000	1.000+000	1.599+001
2	N158(N,P)CO58	1.132-007	1.000+000	1.504+000	1.000+000	1.427+000
3	CU63(N,A)CO60	4.170-009	1.000+000	2.358+000	1.000+000	1.527+000
4	TI46(N,P)TI46	9.568-008	1.000+000	1.577+000	1.000+000	9.955+000
5	CO59(N,G)CO60	4.167-009	1.000+000	2.359+000	1.000+000	9.786-001
6	FE58(N,G)FE59	1.799-007	1.000+000	1.346+000	1.000+000	2.810+002
7	U238(N,F)CS137	7.279-010	6.000-002	6.196+000	1.000+000	3.953+000
8	U238(N,F)ZR95	1.246-007	5.105-002	1.467+000	1.000+000	3.953+000
9	U238(N,F)RU103	2.035-007	6.229-002	1.312+000	1.000+000	3.953+000
10	U238(N,F)CE141	2.468-007	5.456-002	1.265+000	1.000+000	3.953+000
11	U238(N,F)RU106	2.180-008	2.513-002	2.274+000	1.000+000	3.953+000
12	NP237(N,F)CS137	7.279-010	6.267-002	6.196+000	1.000+000	3.936+000
13	NP237(N,F)ZR95	1.246-007	5.699-002	1.467+000	1.000+000	3.936+000
14	NP237(N,F)RU103	2.035-007	5.584-002	1.312+000	1.000+000	3.936+000
15	NP237(N,F)CE141	2.468-007	5.439-002	1.265+000	1.000+000	3.936+000
16	NP237(N,F)RU106	2.180-008	2.240-002	2.274+000	1.000+000	3.936+000
17	NP237(N,F)CE144	2.821-008	4.198-002	2.216+000	1.000+000	3.936+000
18	ZR94(N,G)ZR95	1.252-007	1.000+000	1.465+000	1.000+000	8.706+000
19	SN124(N,G)SN125	7.935-009	1.000+000	2.226+000	1.000+000	3.520+001
20	U238(N,F)CE144	2.821-008	4.543-002	2.216+000	1.000+000	3.953+000

* UNIT CONVERSION FACTOR FROM DPS/MG TO DPS/ATOM

REACTION		EOI RATE*	SATURATION	GRADIENT	IMPURITY	REACTION RATE	% UNCERT **
NAME	IDENT	(DPS/MUCL)	FACTOR	FACTOR	FACTOR	(RPS/MUCL)	(1 STD)
FE54(N,P)FM54	Fe 1	3.70-017	2.243+000	1.000	1.000	8.29-017	
	Fe 2	4.86-016	2.243+000	1.000	1.000	1.09-015	
	Fe 3	7.21-016	2.243+000	1.000	1.000	1.62-015	
	Fe 333	1.17-015	2.243+000	1.000	1.000	2.63-015	
	Fe 44	2.56-014	2.243+000	1.000	1.000	5.75-014	
	Fe 4	4.58-013	2.243+000	1.000	1.000	1.03-012	
	Fe 6	6.54-013	2.243+000	1.000	1.000	1.47-012	
	Fe 7	7.82-013	2.243+000	1.000	1.000	1.75-012	
	Fe 777	8.21-013	2.243+000	1.000	1.000	1.84-012	
	Fe 88A	6.64-013	2.243+000	1.000	1.000	1.49-012	
	Fe 88B	6.74-013	2.243+000	1.000	1.000	1.51-012	
	Fe AA	8.36-013	2.243+000	1.000	1.000	1.87-012	
	Fe JJ	5.50-013	2.243+000	1.000	1.000	1.23-012	
	Fe X	2.52-013	2.243+000	1.000	1.000	5.65-013	
CO59(N,G)CO60	Co 1	2.77-020	2.359+000	1.000	1.000	6.53-020	
	Co 2	5.82-019	2.359+000	1.000	1.000	1.37-018	
	Co 3	7.45-019	2.359+000	1.000	1.000	1.76-018	
	Co 333	3.88-018	2.359+000	1.000	1.000	9.14-018	
	Co 44	3.95-016	2.359+000	1.000	1.000	9.33-016	
	Co 4	1.32-015	2.359+000	1.000	1.000	3.12-015	
	Co 6	1.83-015	2.359+000	1.000	1.000	4.32-015	
	Co 7	8.01-016	2.359+000	1.000	1.000	1.89-015	
	Co 777	7.66-016	2.359+000	1.000	1.000	1.81-015	
	Co 88A	8.55-016	2.359+000	1.000	1.000	2.02-015	
	Co 88B	7.72-016	2.359+000	1.000	1.000	1.82-015	
	Co AA	6.89-016	2.359+000	1.000	1.000	1.63-015	
	Co JJ	6.95-016	2.359+000	1.000	1.000	1.64-015	
	Co X	1.32-015	2.359+000	1.000	1.000	3.12-015	
ZR94(N,G)ZR95	Zr 777	2.85-012	1.465+000	1.000	1.000	4.18-012	
SN124(N,G)SN125	Sn 777	2.20-011	2.226+000	1.000	1.000	4.89-011	

* End-of-irradiation (EOI) specific activity.

** Based on 1) the PT 2954 and PT 3053 analyses (2) and 2) a review of the information in References [1,5,6], the counting statistics are assumed to be in the range of 5 to 20% (one-sigma). Similarly, uncertainties in the measured content (weight) of Fe54, Zr94, and Sn124 are assumed to be in the 5 to 10% (one-sigma) range.

[Ref. 1] Trimble 1988.

[Ref. 5] Marshall and Scott 1984.

[Ref. 6] Lowry et al. 1985.

APPENDIX G

N-REACTOR PRESSURE TUBE 1165 PHYSICS-DOSIMETRY RESULTS

APPENDIX G

N-REACTOR PRESSURE TUBE 1165 PHYSICS-DOSIMETRY RESULTS

The following is an excerpt from PNL-NDC Letter Report 3, "FERRET-SAND II Physics-Dosimetry Analysis for the N-Reactor Pressure Tube 1165 Using a WIMS Calculated Input Spectrum," by W. N. McElroy, L. S. Kellogg, and A. E. Suski, dated March 28, 1988.

1.0 Objective and Workscope

Objective : To provide TIMWH and FERRET-SAND II physics-dosimetry results for N-Reactor Pressure Tube (PT) 1165 to permit comparison of methods used by Battelle Columbus Laboratory (BCL) to those used by Battelle Pacific Northwest Laboratories (PNL) for calculating fast fluence ($E > 1\text{Mev}$) from measured specific activities of Mn-54 in pressure tube samples.

Workscope : Calculate fast fluences ($E > 1\text{Mev}$) for PT 1165. These calculations are to be as similar as possible to those previously made by the PNL-National Dosimetry Center (NDC) in 1987 for Pressure Tubes 2954 and 3053. These calculations are to use Westinghouse Hanford Company (WHC) supplied information as follows : a) the BCL specified iron contents and BCL (August 1983) measured specific activities for the Mn-54 reaction from 14 PT 1165 samples; b) the tube's tabulated reactor power time history data; and c) the same WIMS 69 group input flux-spectrum and the same iron burn-out factor as used for the Tubes 2954 and 3053 evaluations.

2.0 Summary

TIMWH and FERRET-SAND II physics-dosimetry results for N-Reactor Pressure Tube (PT) 1165 have been determined in order to compare methods used by Battelle Columbus Laboratory (BCL) to those used by Battelle Pacific Northwest Laboratories (PNL) National Dosimetry Center (NDC) for calculating fast fluence ($E > 1\text{Mev}$) from measured specific activities of Mn-54 in 14 axial pressure tube samples.

Fe54(n,p)Mn54 reaction rates (Mn54 saturated activities) for 14 axial locations have been derived using the TIMWH Code; see Table 1. These reaction rate data were then used together with other FERRET-SAND II Code input data to derive a complete set of neutron exposure and exposure rate parameter values with uncertainties for the N-Reactor PT 1165 Sample 777 axial location (325.2 inches)*. Values for flux and fluence [total ($1.0\text{E}-10 < E < 18\text{ MeV}$); thermal ($E < 0.414\text{ eV}$); fast ($E > 0.1$ and 1.0 MeV)]; dpa/sec and dpa in iron; and a comparison of measured and calculated reaction rates are presented in tabular form for this one location; see Tables 2 and 3. Table 1 and Figure 1 provide the FERRET-SAND II derived results on the axial variation of fast fluence ($E > 0.1, 0.18,$ and 1 MeV) values for the 14 sample locations. These exposure parameter values were obtained by using the FERRET-SAND II derived values of the Fe54(n,p)Mn54 spectral averaged cross section for ($E > 0.1, 0.18,$ and 1 MeV) and the Table 1 TIMWH derived values of reaction rates.

The PNL-NDC and BCL derived Fe54(n,p)Mn54 reaction rates using the TIMWH and Decay Codes, respectively, show differences of up to about 12.8%. The PNL-NDC derived values of fast flux and fluence ($E > 1\text{Mev}$) are higher (by up to a factor of 1.36) than the BCL reported values for the PT 1165 Sample 777 location; see Table 4.

The largest difference in the two BCL and one PNL-NDC reported exposure parameter values is traceable to the use of significantly different values of the Fe54(n,p)Mn54 reaction's spectral averaged cross section [$\bar{\sigma}$ ($E > 1\text{Mev}$)] for flux and fluence ($E > 1\text{Mev}$). The current BCL and PNL-NDC derived values of $\bar{\sigma}$ ($E > 1\text{Mev}$) are based on the use of WHC supplied flux-spectra that have not been adjusted by dosimetry measurements. These PT 1165, as well as previous surveillance program, results emphasize, therefore, the importance of the derivation and use of dosimetry adjusted values of spectral-averaged reaction cross sections for the determination of reliable (± 5 to 20%, one-sigma) exposure parameter values.

* As measured from the pressure tube inlet end.

The following is an excerpt from PNL-NDC Letter Report 4, "Fluence Measurements Investigations for N Reactor Pressure Tube 1165," by W. N. McElroy, L. S. Kellogg, W. C. Morgan, and A. E. Suski, dated May 9, 1988.

1.0 Objective and Workscope

Objective :

To evaluate uncertainties in calculations of neutron flux-fluence exposures obtained from measured specific activities of dosimetry samples taken from N-Reactor pressure tubes.

Workscope :

1. Using FERRET-SAND II, calculate the $\text{Fe54}(n,p)\text{Mn54}$ ($E > 1$ MeV) spectrum averaged cross section using two input flux-spectra; i.e., a 48 group WIMS and a 92 group MCNP based interpolated-extrapolated multi-group flux-spectrum.
2. Compare calculational results from 1) to values previously calculated by McElroy et al. using a 69 group WIMS input flux-spectrum.
3. Recommend a best estimate of the $\text{Fe54}(n,p)\text{Mn54}$ ($E > 1$ MeV) spectrum averaged cross section.
4. Evaluate the cause(s) for the differences between the PNL-NDC and BCL derived values of the $\text{Fe54}(n,p)\text{Mn54}$ reaction rate obtained from the BCL measured specific activity of Pressure Tube 1165 Sample 777.

2.0 Summary

The primary factor responsible for the differences between Battelle Pacific Northwest Laboratories-National Dosimetry Center (PNL-NDC) and BCL neutron fluxes and fluences ($E > 1$ MeV) for Pressure Tube 1165 Sample 777 is associated with the differences in the derived values of the $\text{Fe54}(n,p)\text{Mn54}$ spectrum averaged cross section. Other factors are identified in decreasing order of importance.

It has been determined that the 48 group flux-spectrum provided to BCL by WHC (UNC) is incorrect, i.e., the numerical manipulations used to collapse the 69 group to the 48 group WIMS flux-spectrum were done incorrectly.

The best estimate of the $\text{Fe54}(n,p)\text{Mn54}$ spectral averaged cross section value [$\bar{\sigma}$ ($E > 1$ MeV)] for use in the derivation of N-Reactor pressure tube fluxes and fluences ($E > 1$ MeV) is 86.2 mb. This value is the average of the two values derived with the FERRET-SAND II code using the 69 group WIMS and the 92 group MCNP based flux-spectra as input.

The results of a N-Reactor pressure tube physics-dosimetry experiment would be required to determine a more accurate value of $\bar{\sigma}$ ($E > 1$ MeV).

TABLE 1

ESTIMATED FLUENCES FOR PRESSURE TUBE 1165

PROCESS TUBE SPECIMEN IDENT	DISTANCE FROM END OF TUBE (INCHES)	MEASURED REACTION RATE (RPS/ATOM)	BASED ON THE MEASURED FE54(N,P)MN54 REACTION RATE*		
			FLUENCE	FLUENCE	FLUENCE
			E>0.1MeV (n/cm**2)	E>0.18MeV (n/cm**2)	E>1MeV (n/cm**2)
Fe 1	38.4	8.29E-17	6.57E+17	5.71E+17	2.48E+17
Fe 2	51.6	1.09E-15	8.64E+18	7.51E+18	3.26E+18
Fe 3	84.0	1.62E-15	1.28E+19	1.12E+19	4.85E+18
Fe 333	104.4	2.63E-15	2.08E+19	1.81E+19	7.87E+18
Fe 44	116.4	5.75E-14	4.56E+20	3.96E+20	1.72E+20
Fe 4	171.6	1.03E-12	8.17E+21	7.10E+21	3.08E+21
Fe 6	216.0	1.47E-12	1.17E+22	1.01E+22	4.40E+21
Fe 7	282.0	1.75E-12	1.39E+22	1.21E+22	5.24E+21
Fe 777	325.2	1.84E-12	1.46E+22	1.27E+22	5.51E+21
Fe 88A	393.6	1.49E-12	1.18E+22	1.03E+22	4.46E+21
Fe 88B	393.6	1.51E-12	1.20E+22	1.04E+22	4.52E+21
Fe AA	441.6	1.87E-12	1.48E+22	1.29E+22	5.60E+21
Fe JJ	490.8	1.23E-12	9.75E+21	8.48E+21	3.68E+21
Fe X	549.6	5.65E-13	4.48E+21	3.89E+21	1.69E+21

* The column 3 measured values were increased by a factor of 1.048 to account for the burn-out of Fe54 target atoms over approximately 18 years of reactor operation; for the present study, however, no axial correction has been made for the drop-off of this central tube region burn-out factor with decreasing flux near and at the ends of the process tube. Consequently, the columns 4, 5 and 6 derived values of fluence at and near the ends of the process tubes are too high, but only by a very small percentage. The 1.048 factor was determined for 23 years of reactor operation for Pressure Tubes 2954 and 3053 and, therefore, is slightly too high for Pressure Tube 1165.

TABLE 2
FERRET-SANDII RESULTS FOR PROCESS TUBE 1165

GROUP	ENERGY ^(a) (MEV)	A PRIORI FLUX* (N/CM**2/SEC)	ADJUSTED FLUX (N/CM**2/SEC)	% UNCERTAIN (1 STD)
1	1.492+001	1.006+009	1.006+009	52.
2	1.350+001	2.092+009	2.092+009	49.
3	1.162+001	9.700+009	9.699+009	45.
4	1.000+001	2.944+010	2.944+010	41.
5	8.607+000	7.448+010	7.447+010	37.
6	7.408+000	1.607+011	1.607+011	32.
7	6.065+000	4.454+011	4.454+011	27.
8	4.966+000	7.414+011	7.414+011	24.
9	3.679+000	1.713+012	1.713+012	20.
10	2.865+000	2.306+012	2.306+012	22.
11	2.231+000	3.043+012	3.043+012	27.
12	1.738+000	3.554+012	3.554+012	32.
13	1.353+000	3.681+012	3.681+012	37.
14	1.108+000	2.946+012	2.946+012	41.
15	8.208-001	4.772+012	4.772+012	45.
16	6.393-001	4.838+012	4.838+012	49.
17	4.979-001	4.519+012	4.519+012	52.
18	3.877-001	3.561+012	3.561+012	55.
19	3.020-001	3.508+012	3.508+012	57.
20	1.832-001	6.912+012	6.912+012	59.
21	1.111-001	5.947+012	5.947+012	61.
22	6.738-002	5.202+012	5.202+012	62.
23	4.087-002	4.867+012	4.867+012	63.
24	2.554-002	4.369+012	4.369+012	63.
25	1.989-002	2.296+012	2.296+012	64.
26	1.503-002	2.545+012	2.545+012	64.
27	9.119-003	4.213+012	4.213+012	64.
28	5.531-003	3.761+012	3.761+012	64.
29	3.355-003	4.260+012	4.260+012	65.
30	2.839-003	1.308+012	1.308+012	65.
31	2.404-003	1.297+012	1.297+012	65.
32	2.035-003	1.292+012	1.292+012	65.
33	1.234-003	3.869+012	3.869+012	65.
34	7.485-004	3.862+012	3.862+012	65.
35	4.540-004	3.795+012	3.795+012	65.
36	2.754-004	3.667+012	3.667+012	65.
37	1.670-004	3.622+012	3.622+012	65.
38	1.013-004	3.618+012	3.618+012	65.
39	6.144-005	3.586+012	3.586+012	65.
40	3.727-005	3.542+012	3.541+012	65.

a) Lower Energy Bound

TABLE 2 (Cont'd)

FERRET-SANDII RESULTS FOR PROCESS TUBE 1165

GROUP	ENERGY (MEV)	A PRIORI FLUX* (N/CM**2/SEC)	ADJUSTED FLUX (N/CM**2/SEC)	% UNCERTAIN (1 STD)
41	2.260-005	3.448+012	3.447+012	65.
42	1.371-005	3.344+012	3.344+012	65.
43	8.315-006	3.258+012	3.258+012	65.
44	5.043-006	3.101+012	3.101+012	65.
45	3.059-006	3.141+012	3.140+012	65.
46	1.855-006	3.261+012	3.261+012	65.
47	1.125-006	3.363+012	3.363+012	65.
48	6.826-007	3.592+012	3.592+012	65.
49	4.140-007	4.618+012	4.618+012	65.
50	2.511-007	8.547+012	8.546+012	65.
51	1.523-007	1.523+013	1.523+013	65.
52	9.237-008	1.798+013	1.798+013	65.
53	1.000-010	3.095+013	3.095+013	65.

FLUX, E> 0.0 MEV		2.216+014	2.216+014	43. %
FLUX, E< 0.414 EV		7.270+013	7.270+013	61. %
FLUX, E> 0.1 MEV		5.386+013	5.386+013	37. %
FLUX, E> 1.0 MEV		2.034+013	2.034+013	21. %
DPA/SECOND		3.186-008	3.186-008	24. %
FLUENCE+, E> 1.0 MEV		5.514+021	5.514+021	21. %
DPA+		8.637+000	8.637+000	24. %

* ABSOLUTE CALCULATED FLUX (OR NORMALIZED TO FE54(N,P)MN54 REACTION RATE).

+ BASED ON 2.711+008 EFFECTIVE 4.07 MW TUBE POWER SECONDS OF EXPOSURE; UNITS OF N/CM**2 FOR FLUENCE AND TOTAL NUMBER OF DISPLACED ATOMS OF IRON PER TARGET ATOM OF IRON FOR DPA.

TABLE 3

COMPARISON OF MEASURED AND CALCULATED REACTION RATES
USED IN THE ANALYSIS OF PRESSURE TUBE 1165 (325.2 inches)

REACTION	COVER	REACTION RATE (DPS/NUCLEUS)				RATIO MEAS/CALC	
		-----				-----	
		(a)				(b)	
		MEAS	A PRIORI	CALC	ADJ CALC	A PRIORI	ADJ CALC
FE54(N,P)MN54	BARE	1.93E-12(8)	1.93E-12		1.93E-12	1.00	1.00

- a) The measured value of 1.84E-12, Table 1, was multiplied by the Fe54 target burnout correction factor of 1.048. The bracketed value is the one-sigma(%) input uncertainty estimate used for this FERRET-SANDII run.
- b) The relative WIMS 69 group input flux-spectrum was normalized to the Fe54(n,p)Mn54 measured reaction rate so that the a priori ratio of meas/calc was unity.

TABLE 4

COMPARISON OF BCL AND PNL-NDC
FERRET-SANDII RESULTS FOR PROCESS TUBE 1165 SAMPLE 777 LOCATION

INTEGRAL FLUX(a) AND FLUENCE VALUES	BCL FLUX AND FLUENCE VALUES (Ref. 5) (Ref. 6)		PNL-NDC FLUX & FLUENCE VALUES(b)	FERRET-SANDII % UNCERTAINTY (1 STD)
FLUX, E> 0.0 MEV	-	-	2.22E+014	43. %
FLUX, E< 0.414 EV	-	-	7.27E+013	61. %
FLUX, E> 0.1 MEV	-	-	5.39E+013	37. %
FLUX, E> 0.18 MEV	-	-	4.68E+013	< 37. %
FLUX, E> 1.0 MEV	1.80E+013	1.56E+013	2.03E+013	21. %
FLUENCE, E>1 MEV	4.69E+021	4.04E+021	5.51E+021	21. %
FLUX RATIOS(c)	1.15	1.000	1.30	
FLUENCES RATIOS(c)	1.16	1.000	1.36	

- a) Units of n/cm²-sec. for flux and n/cm² for fluence.
- b) Tables 1 and 2 FERRET-SANDII results.
- c) Normalized to the BCL (Ref. 6) flux and fluence values.
- (Ref. 5) Marshall and Scott 1984.
- (Ref. 6) Lowry et al. 1985.

FIGURE 1

PRESSURE TUBE 1165 FLUENCE PROFILES

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