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SUBJECT: Forwards addl info re reactor vessel matl surveillance program, analysis of Capsule V, per 861121 request.

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FEBRUARY 26 1987
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U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Gentlemen:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Reactor Vessel Material Surveillance Program
Request for Additional Information
NRC TAC Nos. 62760 and 62761

Attached is our response to your November 21, 1986 request for additional information regarding the Reactor Vessel Material Surveillance Program for Turkey Point Unit 3, analysis of Capsule V.

If you have any further questions, please call us.

Very truly yours,

J. A. DeMastry

C. O. Woody for
Group Vice President
Nuclear Energy

Attachment

COW/TCG/cvb

cc: Dr. J. Nelson Grace, Regional Administrator, NRC Region II
Mr. D. R. Brewer, Sr. Resident Inspector, Turkey Point Plant

A001 1/1

DDI/NRC/I

ANSWERS TO NRC QUESTIONS ON REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM ANALYSIS OF CAPSULE V TURKEY POINT UNIT 3

Question 1: An experimental error analysis should be performed to support vessel fluence uncertainty values.

Response:

The estimated experimental error provided relates directly to the activity of the dosimeter at the time of removal (A_{tor}) quoted in the revised Table X (see Response 2) of the submitted report "Reactor Vessel Material Surveillance Program for Turkey Point Unit 3; Analysis of Capsule V", SRI Project 06-8575.

The gamma ray emission rates in disintegrations per second (DPS) were determined with a germanium detector with an IT-5400 multichannel analyzer according to Southwest Research Institute (SRI) procedures which reference ASTM standards appropriate to processing dosimetry. In general, these standards are designed to yield gamma ray emission rates with an uncertainty of + 3% at the 68% confidence level. For the specific case of the determination of A_{tor} for Capsule V, the following three sources of error were identified along with an estimate of their magnitude:

a) Random counting error = ± 3%
b) Systematic counting hardware error = ± 5%
c) Calibration source error = ± 2%

These errors are independent and statistically combine to yield a total error on the order of 6% (1s).

The weight of the dosimeter used in the determination of the activity per milligram (DPS/mg) was established with a Mettler balance with a quoted accuracy of ± .1 microgram. Given the range of dosimeter weights in Table X (8.4 to 2107 milligram) the error in A_{tor} due to uncertainnty in the dosimeter weight alone is less than 1% and is not considered a significant contributor to the experimental error in A_{tor}.

For the purpose of interpretation of the measurement results relative to calculational results, a total measurement error of 10% is considered to be an upper bound in view of the use of the information quoted by accepted industry standards appropriate to the determination of dosimeter activity and the specific estimates for Capsule V.

Question 2: There is no justification for the rejection of the dosimeter measurements and the exclusive use of dosimeters from the Charpy bars.

Response:

The neutron dosimetry in the surveillance Capsule V contained three neutron dosimeter capsules. They were positioned vertically at top, middle and bottom of the compartments of the surveillance Capsule V. Each dosimeter capsule contained copper, nickel, al-cobalt wire (cadmium-covered and uncovered), Np-237 and U-238 threshold dosimeters. Also, the Charpy test specimens served as iron threshold dosimeters. Only copper, al-cobalt, and iron dosimeters provide reliable neutron fluence measurement data. All others are regarded with suspicion. The questionable measurement data given in the revised Table X of the SRI report, were not used for comparison between measured and calculated fluence data. Justifications for the rejection of the suspected dosimeters are given below:

1. Both uranium and neptunium dosimeters were powder specimens and were contained in metal capsules of either brass or steel. During the opening of the metal capsules, both dosimeters were contaminated by metal filings from the sawing of these metal capsules. The contamination increased the weights of uranium and neptunium dosimeters. Thus, the dosimeter's specific activity in (DPS/mg) became questionable due to uncertainties in the actual weights.

2. The basis for rejection of the nickel dosimeter which generates the Co-58 gamma emitter is that no expected Co-58 photo-peak was observed from the gamma ray spectroscopy counting. Instead, two 1.17 and 1.33 MeV photo-peaks characteristic of the Co-60 isotope were observed from the counting. It is believed that a Co-59 wire rather than Ni wire was loaded into the dosimeter capsule. Therefore, the intended nickel dosimeter as an integrated fluence indicator was not available.

The revised Table X reflects two corrections from the old Table X.

a. A correction for using old NBS standard source Co-60 to analyze the gamma counting data.

b. A minor correction for iron weight percentage in the Charpy test specimens.

It should be pointed out that measurement data from iron and copper dosimeters was used for comparisons.

Question 3: A computational error analysis should be performed to evaluate the relative value of the computed to the measured fluence.

Response:

The error analysis provided is an estimate of the overall computational error resulting from the use of nominal input data which is subject to variability or uncertainty about its nominal value. In this analysis two reactor vessel flux calculation reference cases were established: The first reference case is a one-dimensional (1-D) transport code - ANISN case, and the second reference case is a two-dimensional (2-D) transport code -DOT4.3 case. The two reference cases utilized the nominal values of the analytical model input data.

To obtain a deviation of the nominal value of input data, a reasonable data uncertainty range was assigned to the nominal value. An additional reactor vessel flux calculation with new input data (nominal value $\pm$ data uncertainty) was performed to determine new vessel flux data.

A comparison of the new calculated and reference calculated vessel fluxes was made to obtain the computational vessel flux data uncertainty.

Figure 1 shows a standard vessel flux calculational flow-chart with alphabetic labels for identification of the input data uncertainty studied. Table 1 lists each alphabetic label associated with the actual input data uncertainty. The last column of Table 1 indicates the computed vessel flux variation over the reference vessel flux data. An overall computational vessel flux error was found to be $\pm$ 27.5%. Table 2 presents conditions for the two reference cases. It should be pointed out that the above vessel flux uncertainty analysis was based on a single cycle burnup average core power data. Since analytical methodologies for vessel flux calculations of different fuel cycles are identical, the calculated $\pm$ 27.5% vessel flux error is applicable to cycles 1 through 9.

The relative value of the computed to measured fluence for Capsule V was determined to be .88. This was established by an independent fluence analysis performed by FPL. Table 3 presents a comparison of measured and calculated neutron ($E \geq 1.0$ MeV) fluence results for the Turkey Point Units 3/4 since 1975 up to present. The measurement data given in Table 3 are from the in-vessel and ex-vessel neutron dosimetry program at FPL and the calculational data are from the FPL diffusion (PDQ-7) + transport (DOT4.3) computer codes package. Table 3 indicates a consistent underprediction of the measurements. The uncertainties and the

consistent calculational bias identified above provide a sound basis for the conclusions drawn in Response 4 regarding the adequacy of the Capsule V measurement data.

<u>Question 4:</u> Justify the use of the P_1 approximation.

<u>Response:</u>

An independent Capsule V neutron fluence analysis using the industry accepted P_3 approximation was performed by FPL for the purpose of evaluating the measurement results reported by SRI, which relied on a lower order P_1 cross-section approximation.

The cumulative fluence results reported by SRI were derived from several factors of which only the radial flux gradient correction factor inside Capsule V and the effective spectrum-averaged dosimeter reaction cross-section depend upon the P_1 approximation. The other factors are either dependent upon direct measurement or industry accepted constants.

Figure 2 shows the neutron flux distributions resulting from the P_1 and P_3 cross-section approximation analyses. The effect of the higher order cross-section approximation becomes evident in the capsule region. A 1.2% flux increase is observed with respect to the P_1 cross-section approximation case. However, the radial flux gradients inside the capsule region are almost the same for both the P_1 and P_3 cases. The reason for this is that the surveillance capsule is sufficiently far from the source of neutrons that the bias between the P_1 and P_3 approximation for all practical purposes is constant in the Capsule V region.

As for the determination of the effective cycle-specific spectrum-weighted average dosimeter reaction cross section, FPL established that this value did not significantly vary over each of the nine operating cycles and was very similar to the value established by SRI.

Table 4 lists the FPL calculated 9-cycle average spectrum-weighted reaction cross-sections for iron and copper dosimeters. Also provided in Table 4 are the SRI reported cross-sections for comparison. At most, a 3.0% cross-section difference is noticed between the P_1 and P_3 cross-section approximation cases.

The reason that the calculation of this factor is not strongly influenced by the order of the scattering approximation used is that it is a spectrum weighted average defined as:

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

in which biases due to the order of the scattering approximation in the calculated energy dependent flux $\phi(E)$ tend to cancel.

The reasons provided above, as substantiated by the FPL performed P_3 calculation, justify the use of the P_1 approximation in the SRI reported measurement results.

The calculated fluence at the Capsule V location for each of the nine operating cycles is presented in Table 5.

As indicated in Table 3 of the preceding response, Capsule V actually received the cumulative neutron ($E \geq 1.0$ MeV) fluence of 1.25E+19 (neutrons/cm^2) for plant operation of cycle 1 through cycle 9. The corresponding calculated Capsule V neutron fluence is 1.104E+19 (neutrons/cm^2) for plant operation of cycle 1 through cycle 9. The ratio of calculated to measured neutron fluences is 0.88 which shows a 12% underprediction of measurement.

The uncertainty in the calculated neutron fluences due to uncertainties in the analytical input data and models is estimated to be $\pm$ 27.5% as given in Table 1 of the preceding response. The uncertainty in measured neutron fluence at Capsule V is estimated to be $\pm$ 10% and is shown in the Response to Question 1. If a 12% of underprediction of measurement is used in the calculated neutron fluence data, the variation of measured neutron fluence at Capsule V is between +2% and 22% which is well enveloped by the variation of the calculated neutron fluences from the range of $\pm$ 27.5%.

This provides additional evidence that the reported measurement results (with the exceptions noted in Response 2) are accurate.

Question #5 Explain the counting in dosimeters which are not chemically pure.

Response:

In Capsule V, only the iron dosimeters made from the irradiated Charpy test specimens were not chemically pure. Chemistry of the Charpy test specimens are typically 97% iron and 0.68% nickel, with the remainder being other alloy materials. Due to neutron activation of the Charpy test specimens during its residence inside the reactor vessel, two prominent gamma emitters were produced. They are the Mn-54, product of the Fe54(N,P)Mn54 reaction, and the Co-58, product of the Ni58(N,P)Co58 reaction. The Mn-54 gamma emitter has a half-life of 321.5 days and emits a 835 KeV photon. The Co-58 gamma emitter has a half-life of 71 days and emits a 811 KeV photon. However, only the iron activation product Mn-54 gamma counting is of interest for dosimetry purposes. Since no chemical separation of nickel from iron was done, the nickel activation product Co-58 gamma can potentially interfere with the Mn-54 gamma counting due to the proximity of the Co-58 photo peak.

In the gamma counting procedure, an IT-5400 multichannel analyzer and a conventional Ge(Li) coaxial detector were used to measure the gamma activity of the Mn-54. Before measurements, the counting system was calibrated by using standard test gamma sources obtained from the National Bureau of Standards. A typical 0.5 KeV per channel of IT-5400 multichannel analyzer and 0.25 percent of energy resolution (at 835 KeV) of the counting system were used for measurements. There were 48 channels of separation between the Mn-54 and the Co-58 photo peaks (48 channels = (835-811 KeV)/0.5 KeV/channel). 0.25 percent energy resolution provided the estimated 2-KeV full width at half maximum (FWHM) spreadings for both gammas. This in turn inferred that there were 44 channels (44x0.5 KeV) of separation between the Mn-54 and Co-58 photon distributions. At the time of measurement, at least 215 days had expired since Capsule V was removed from the reactor. No significant interference of the Mn-54 photo peak due to the proximity of the Co-58 photo peak is expected for the following reasons:

a. the lower production rate of the Co-58 gamma from nickel impurity at the end of neutron irradiation

b. at the time of counting, the Co-58 emitter had significantly decayed as a result of the shorter half-life (71 days) of Co-58 as compared with the half-life (312.5 days) of Mn-54

c. high energy resolution of the counting system

An analytical evaluation was used to estimate the magnitudes of the total gamma counting of the Mn-54 and the Co-58 in order to substantiate the above statement. Estimation was made based on the production rate of the Mn-54 and Co-58. Results demonstrate that at the time the measurement was performed, the magnitude of the Mn-54 gamma counting is higher than the Co-58 gamma counting by a factor of 36.

Based on the above results from measurement and analytical estimation, there was no significant distortion of the Mn-54 photon distribution due to the proximity of the Co-58 gamma source.

TABLE 1

Estimated Computational Vessel Flux Uncertainty

Alphabetic Label	Input Data Uncertainty	Estimated Vessel Flux Uncertainty (respect to reference)
A	Downcomer RC ND's variation due to RC temperature changes ± 2°F	± 0.5%
	RC Tave ND variation due to Tave changes ± 4°F	+ 0.4% -0.5%
B	Capsule V positional variation by ± 1.0 cm	-4.0% to +5.0%
C	Core baffle dimension variation ± 1/16"	± 1.0%
	Core barrel dimension variation ± 1/16"	± 1.0%
	Thermal shield dimension variation ± 1/16"	± 1.0%
	Downcomer region (near vessel) dimension variation by 0.5"	+21% to -24.8%
D	Peripheral fuel assembly power variation by 1.05 and 0.95	+3.0% to -3.0%
E	Homogenized core data variation by 1.05 and 0.95	-6.0% to 7.0%
F	1-D ANISN mesh size variation	
	a. In downcomer region	
	1.07 cm/mesh to 0.539 cm/mesh	+0.4%
	1.07 cm/mesh to 2.14 cm/mesh	-0.3%
	b. In thermal shield	
	0.969 cm/mesh to 0.678 cm/mesh	+0.1%
	0.969 cm/mesh to 2.26 cm/mesh	-1.2%
	2-D SORREL (DOT) Mesh Size Variation 0.94°/mesh to 0.82°/mesh	± 0.2%
G	2-D cycle burnup power variation	
	a. BOC Pin Power	+7.1%
	b. EOC Pin Power	~ 0.0%
H	Azimuthal power tilt variation by Tech Spec 1.02 factor applied to core flat fuels	+1.8%
	TOTAL	27.5%

TABLE 2

Reference Case Conditions

Reference Case	Code	Condition
1	ANISN	Standard practice P_3 S_{12} Neutron Source- homogeneous core RC Tave = 575.4°F Boron Concentration - 715 PPM Downcomer RC Temperature - 546.2°F Reactor Internals Dimensions-Nominal Values Reactor Vessel Dimension-Nominal Value Turkey Point Unit 3 Cycle 1 data 27-group cross-sections ($\geq$ 0.1 MeV) Mixed U-235 and Pu239 fission spectra Peripheral assembly radial power gradient
2	DOT4.3	Standard practice P_3S_8. R-Theta Model (One eighth core model) All material surveillance capsules, T,S, and V in calculational models. Azimuthal and radial power gradients- from FPL PDQ-7 pin power files The remaining conditions are the same as reference case 1.

<u>TABLE 3</u>

Comparison of Neutron Dosimetry Data
<u>(From 1975 through 1987)</u>

Unit	Cycle(s)	Dosimeter In-Vessel	Dosimeter Ex-Vessel	Measured Fluence Value	Measured Fluence By	FPL Ratio of Calculated/Measured
3	1	Capsule T		5.68(18)	Westinghouse[1]	0.95
4	1	Capsule T		6.05(18)	SRI[2]	0.92
3	1-9	Capsule V		1.25(19)	SRI[3]	0.88
3	9		12 Measured Data		Westinghouse[4]	0.86

Data Sources:

1. WCAP-8631 Report
2. SRI Project #02-4221 Report
3. SRI Project #06-8575 Report
4. WCAP-11138 Report

TABLE 4

Spectrum-Weighted Average Reaction Cross-Sections for Iron and Copper Dosimeters at Capsule V

Threshold Reaction	$\bar{\sigma}$ (barns) SRI (P_1)	$\bar{\sigma}$ (barns) FPL (P_3)	P_1/P_3
Fe54(N, P)Mn54	0.0786	0.0806	0.98
Cu63(N,α)Co60	0.000885	0.0009088	0.97

TABLE 5

Cycle-Specific Capsule V Neutron ($\geq$ 1.0 MeV)
Fluence for Turkey Point Unit 3

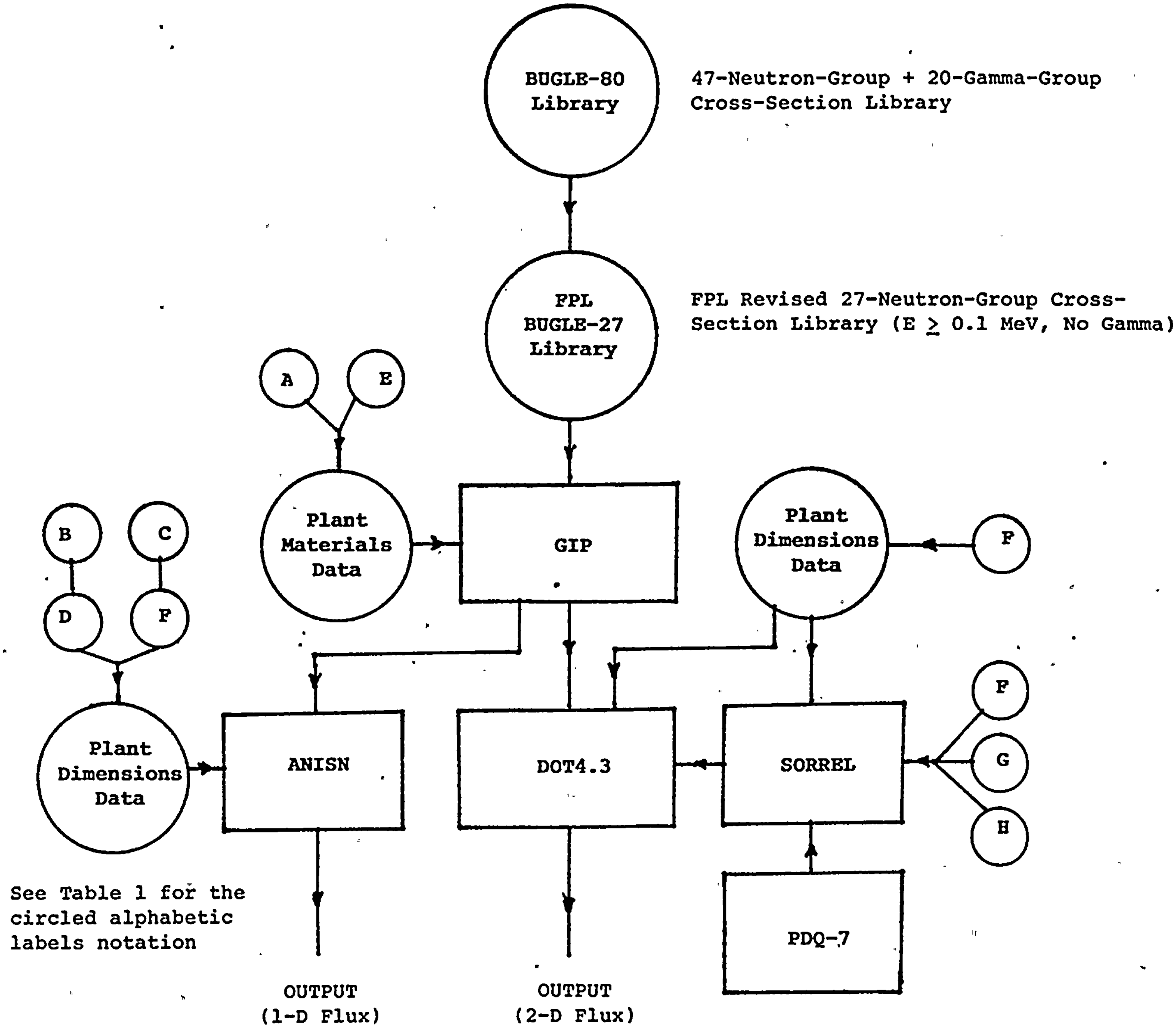
Cycle	Cycle Length (sec)	Charpy Test Specimen Region Average Flux (Neutrons/cm^2-sec)	Cycle Fluence (Neutrons/cm^2)	Cumulative Fluence (Neutrons/cm^2)
1	3.609(7)*	4.596(10)	1.659(18)	1.659(18)
2	2.451(7)	4.574(10)	1.121(18)	2.780(18)
3	2.418(7)	4.908(10)	1.187(18)	3.967(18)
4	2.462(7)	4.356(10)	1.072(18)	5.039(18)
5	2.453(7)	4.498(10)	1.103(18)	6.142(18)
6	1.587(7)	4.325(10)	6.863(17)	6.828(18)
7	2.902(7)	5.397(10)	1.566(18)	8.394(18)
8	4.302(7)	3.612(10)	1.554(18)	9.948(18)
9	3.263(7)**	3.358(10)	1.096(18)	1.104(19)

* Read 3.609(7) as 3.609×10^7

** Actual cycle length rather than planned cycle length has been used in the fluence calculation.

FIGURE 1

CALCULATIONAL FLOW-CHART



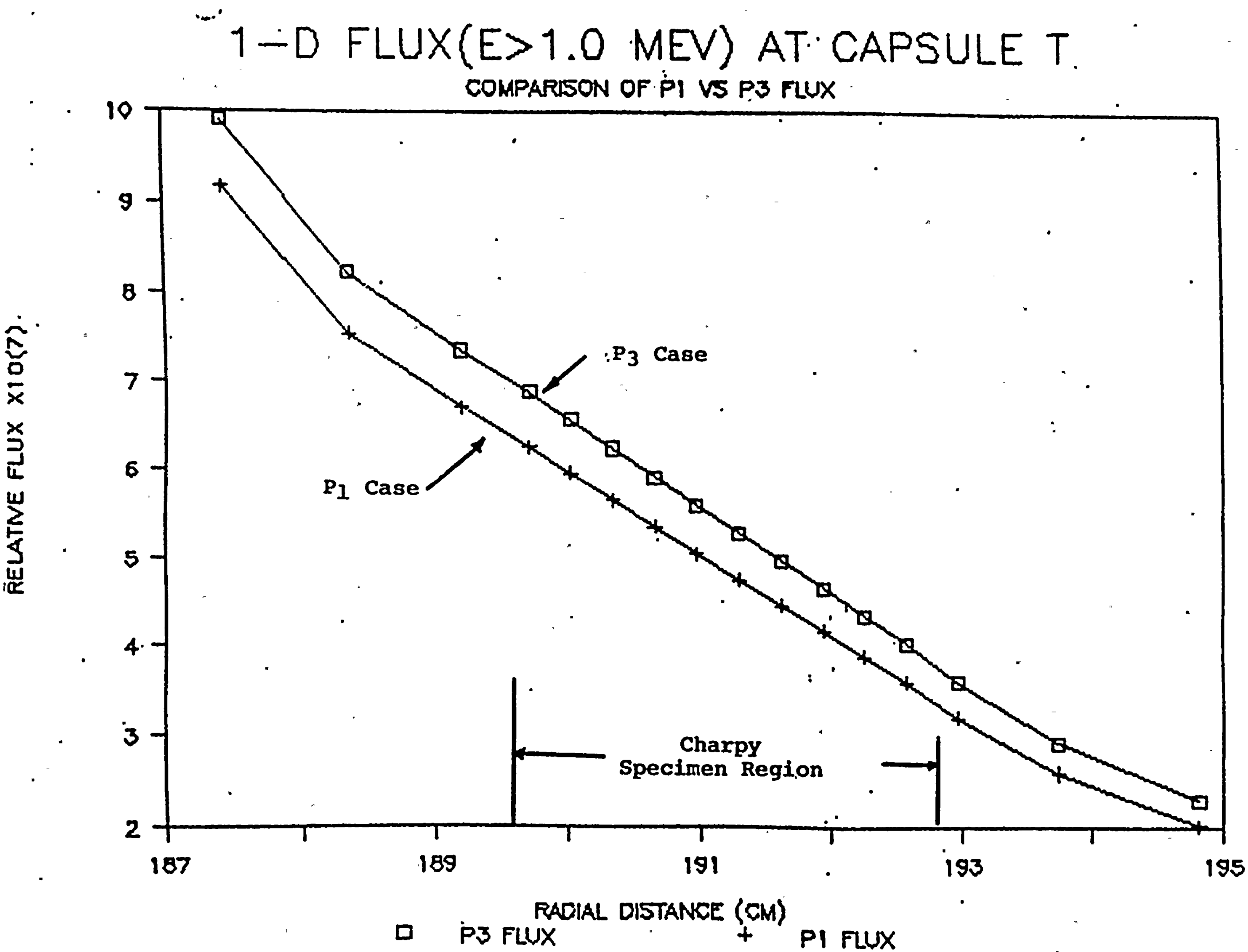


FIGURE 2

Comparison of P_1 and P_3 Fluxes

REVISED TABLE X

February 24, 1987

SATURATED ACTIVITIES AND DERIVED FLUENCE RATES FOR CAPSULE V

Reaction and Axial Location	Radial Location (cm)	Dosimeter Weight (mg)	A_{TOR} (dPS/mg)	A_{SAT} (dPS/atom)	A_{SAT}(a) Adjusted (dPS/atom)	Fluence Rate (a) $n \cdot cm^{-2} \cdot s^{-1}$	Fluence(b) Rate $n \cdot cm^{-2} \cdot s$
$^{54}Fe(n,p)^{54}Mn$							
S-58 TOP	191.47	1839.1	1.96E3	4.21E-15	3.89E-15	5.36E10	4.95E10
S-52 MIDDLE	191.47	2107.3	1.92E3	4.13E-15	3.82E-15	5.25E10	4.86E10
W-2 BOTTOM	191.47	1988.6	1.81E3	3.89E-15	3.59E-15	4.95E10	4.57E10
R-48 TOP	192.47	1955.4	1.63E3	3.50E-15	4.20E-15	4.45E10	5.34E10
R-42 MIDDLE	192.47	1359.5	1.67E3	3.59E-15	4.31E-15	4.57E10	5.48E10
H-2 BOTTOMF	192.47	1827.1	1.52E3	3.27E-15	3.92E-15	4.16E10	4.99E10
$^{63}Cu(n,\alpha)^{60}Co$							
Cu TOP	192.47	156.496	1.22E2	3.53E-17	4.23E-17	3.99E10	4.78E10
Cu BOTTOM	192.47	104.709	1.23E2	3.56E-17	4.26E-17	4.02E10	4.81E10
$^{58}Ni(n,p)^{58}Co$							
Ni MIDDLE	192.47	8.937	No cobalt-58 peak-shows two CO-60 peaks				
$^{237}Np(n,f)^{137}Cs$							
Np MIDDLE	191.69	19.9	2.51E3	9.79E-14	9.79E-14	3.92E10	3.92E10
$^{238}U(n,f)^{137}Cs$							
U MIDDLE	191.69	19.3	2.94E2	1.20E-14	1.20E-14	3.22E10	3.22E10
$^{59}Co(n,\alpha)^{60}Co$ (c)							
Co TOP	191.47	8.474	1.00E7	1.85E-12	2.22E-12	3.03E10	3.03E10
Co-Cd TOP	191.47	9.161	4.93E6	9.13E-13	1.10E-12		
Co MIDDLE	191.47	9.218	9.06E6	1.68E-12	2.02E-12		
Co-Cd MIDDLE	191.47	10.689	No. of counts indicates that count time is 2000 instead of 70,000				
Co BOTTOM	191.47	8.616	9.70E6	1.79E-12	2.15E-12	2.89E10	2.89E10
Co-Cd BOTTOM	191.47	9.434	4.87E6	9.01E-13	1.08E-12		

$A_{SAT} = kA_{TOR}$

(a) At dosimeter location.

(b) Adjusted to radial centerline of specimens.