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FINAL REPORT

on

SURRY UNIT NO. 1 NUCLEAR PLANT REACTOR PRESSURE VESSEL SURVEILLANCE PROGRAM: EXAMINATION AND ANALYSIS OF CAPSULE W

to

VIRGINIA ELECTRIC AND POWER COMPANY

March 30, 1979

by

J. S. Perrin, D. R. Farmelo, R. S. Denning,
R. G. Jung, and E. O. Fromm

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SUMMARY

Capsule W from Surry Unit No. 1 was removed after 3.39 equivalent full power years of reactor operation. The capsule was sent to the Battelle-Columbus Hot Laboratory for examination and evaluation. The maximum irradiation temperature limit was determined by the examination of the three thermal monitors. Capsule and wall fluences were determined by analysis of the neutron dosimeters from the capsule.

The mechanical property specimens, which consist of tensile, Charpy V-notch, and WOL, were not evaluated at this time and were placed in storage at the Hot Lab Facility as requested by VEPCO.

INTRODUCTION

This report presents the results of the examination of the thermal monitors and neutron dosimeters from Capsule W from Surry Unit No. 1. The report contains the experimental procedures, results and discussion relating to the investigation. The mechanical property specimens were not evaluated at this time as directed by VEPCO.

Irradiation of materials such as the pressure vessel steels used in reactors causes changes in the mechanical properties, including tensile, impact, and fracture toughness.[(1-6)*] Tensile properties generally show a decrease of both uniform elongation and reduction in area accompanied by an increase in yield strength and ultimate tensile strength with increasing neutron exposure. The impact properties as determined by the Charpy V-notch impact test generally show a substantial increase in the ductile to brittle transition temperature and a drop in the upper shelf energy.

Commercial nuclear power reactors are put into operation with reactor pressure vessel surveillance programs. The purpose of the surveillance program associated with a reactor is to monitor the changes in mechanical properties as a function of neutron exposure. The surveillance program includes a determination of both the preirradiation base-line mechanical properties and periodic determinations of the irradiated mechanical properties. The materials included in this surveillance program are base metal, weld metal, and heat-affected zone metal. In addition, correlation monitor material specimens are also included.

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* References are listed at the end of the text.

The irradiated mechanical properties are determined periodically by testing specimens from surveillance capsules. These capsules typically contain thermal monitors, neutron flux monitors, Charpy impact specimens, tensile specimens, and fracture mechanics specimens. The capsules are located between the inner wall of the pressure vessel and the reactor core. Capsules are periodically removed, and sent to a hot laboratory for disassembly and specimen evaluation.[7] Surveillance programs are conducted following applicable government and technical society regulations, standards, and codes.[8]

The surveillance program for Surry Unit No. 1 was designed and recommended by the Westinghouse Electric Corporation and is based on ASTM E185, "Surveillance Tests on Structural Materials in Nuclear Reactors".[9] The details of this program and the preirradiation mechanical properties of the materials are presented in Reference 10. Prior to startup, eight capsules containing tensile, Charpy V-notch, and WOL fracture-mechanics specimens of the pressure-vessel materials were installed in the reactor. The capsules were located between the thermal shield and the vessel wall as shown in Figure A-1 of Appendix A. In addition to the mechanical-property test specimens, the capsules contain thermal-monitor and neutron-fluence specimens for evaluation of the temperature upper limit and radiation exposure conditions of the specimens.

The analyses performed in the present program are the fast ($E > 1$ MeV) flux and fluence at the capsule and the maximum temperature experienced by the capsule during the reactor operating period.

CAPSULE RECOVERY AND DISASSEMBLY

Personnel from Battelle's Columbus Laboratory were onsite at the Surry Unit No. 1 Nuclear Plant to provide technical assistance in the loading of surveillance Capsule W in the shipping cask for transport to the Battelle-Columbus Hot Laboratory.

The capsule assembly, which was 112 inches long, was located in the spent fuel storage pit. The capsule lead tube was severed from the capsule and the lead tube was cut into two appropriate lengths to allow placement into the cavity of the shipping cask. The cuts were made using a special cutting tool provided by Battelle.

After the cutting and loading operations were complete, the outside of the cask was decontaminated to levels allowed for shipping. The cask was then shipped from the reactor site to the BCL Hot Laboratory for postirradiation examination.

Upon arrival at BCL, the assembly was removed from the cask and transferred to a hot cell for visual observation, photography, and disassembly. Visual examination showed no unusual features or damage. Figure 1 shows Capsule W in the "as-received" condition at the BCL Hot Laboratory.

The capsule assembly was cut apart using a flexible abrasive wheel attached to a Mototool. During disassembly, the identification number of each specimen was verified against the master inventory list for the specimens in the capsule.[10] Table 1 is an inventory of the test specimens in Capsule W.

FIGURE 1. PHOTOGRAPH OF CAPSULE W FROM SURRY UNIT NO. 1
AS RECEIVED AT THE BCL HOT LABORATORY

TABLE 1. SPECIMEN IDENTIFICATION AND LOCATION IN THE SURRY UNIT NO. 1 IRRADIATION CAPSULE W

Capsule	Specimen Type	Specimen Identification
Top	Dosimeter Thermal Monitor	Co,Co-Cd,Cu C5083 (579 F)
	Charpy	A-29, A-30 V-29, V-30
	Tensile	A-5, A-6
	WOL	A-9, A-8
	Charpy	A-27, A-28 V-27, V-28
	WOL	A-7, V-9
	Charpy	R-23, R-24 R-21, R-22
Capsule	Dosimeter Thermal Monitor	Co,Cd-Co, Ni C5084 (590 F)
	Charpy	R-19, R-20 R-17, R-18
	WOL	V-8, V-7
	Charpy	A-25, A-26 V-25, V-26
	Tensile	V-5, V-6
	Charpy	A-23, A-24 V-23, V-24
Bottom	Charpy	A-21, A-22 V-21, V-22
	Dosimeter Thermal Monitor	Co, Co-Cd, Cu C5082 (579 F)

Specimen Code

A - Plate C 4326-1
V - Plate C 4415-1
R - ASTM Correlation Monitors

There were 28 Charpy, 4 tensiles, and 6 WOL specimens in the capsule. These were placed under oil in marked containers and stored in the Hot Laboratory storage vault as directed by VEPCO.

EXPERIMENTAL PROCEDURES

This section describes the procedures used to evaluate the thermal monitors and neutron dosimeters. All experimental examinations and evaluations were conducted at Battelle's Columbus Laboratories.

Thermal Monitor Examination

The capsule contained two kinds of low-melting-point eutectic alloy thermal monitors for determination of the maximum temperature attained by the test specimens during irradiation. One alloy was 2.5% Ag - 97.5% Pb with a melting point of 579 F. The other alloy was 1.75% Ag - 0.75% Sn - 97.5% Pb with a melting point of 590 F. These thermal monitor alloys were sealed in Pyrex tubes and inserted in spacers in the capsule. During capsule disassembly, the thermal monitors were removed from the spacers and examined for evidence of melting using a stereomicroscope at approximately 5X magnification.

Neutron Dosimetry

The capsule contained a total of 9 dosimeters of copper, nickel, cadmium-shielded aluminum-cobalt alloy, and unshielded aluminum-cobalt alloy. These were in three capsule locations as indicated in Table 1.

In addition, chips from tensile specimens A-6 and V-6 provided iron samples for two additional dosimeters. The reactions used for the dosimetry calculations are as follows.

Iron	^{54}Fe (n,p) ^{54}Mn
Nickel	^{58}Ni (n,p) ^{58}Co
Copper	^{63}Cu (n,α) ^{60}Co
Cobalt	^{59}Co (n,γ) ^{60}Co (Thermal)

All dosimeter samples were recovered and analyzed. After removal from the capsule, the identified samples were placed in individual vials for transfer to the radiochemistry laboratory. The nickel, copper, and Al-Co wires were decontaminated by wiping using successive swabs containing dilute acid, distilled water, and reagent grade acetone. The two tensile specimens for Fe samples were wiped with dilute acid and distilled water to remove major contamination and then cleaned ultrasonically in a solution of Radiac and water until they were completely free of contamination. They were then drilled in one end to obtain a sufficient quantity of chips for analysis.

The iron, nickel, copper, and Al-Co wires were weighed to ±0.0001 g with a calibrated analytical balance, and the activation product intensities were determined by gamma ray spectroscopy.

The activation products were analyzed by utilizing a 3 inch diameter x 3 inch long NaI(Tl) scintillation crystal detector and Model 401D 400 channel analyzer (Technical Measurements Corp.) capable of 7 percent resolution.*

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* Full width-half maximum at the 662 KeV gamma energy level.

The ^{54}Mn and ^{60}Co samples were compared directly with NBS standards. The ^{58}Co activity was obtained from comparison with theoretical efficiency curves prepared from NBS standards.

The procedures used in the evaluation of the dosimetry samples followed the appropriate ASTM recommendations.[11-14]

RESULTS AND DISCUSSION

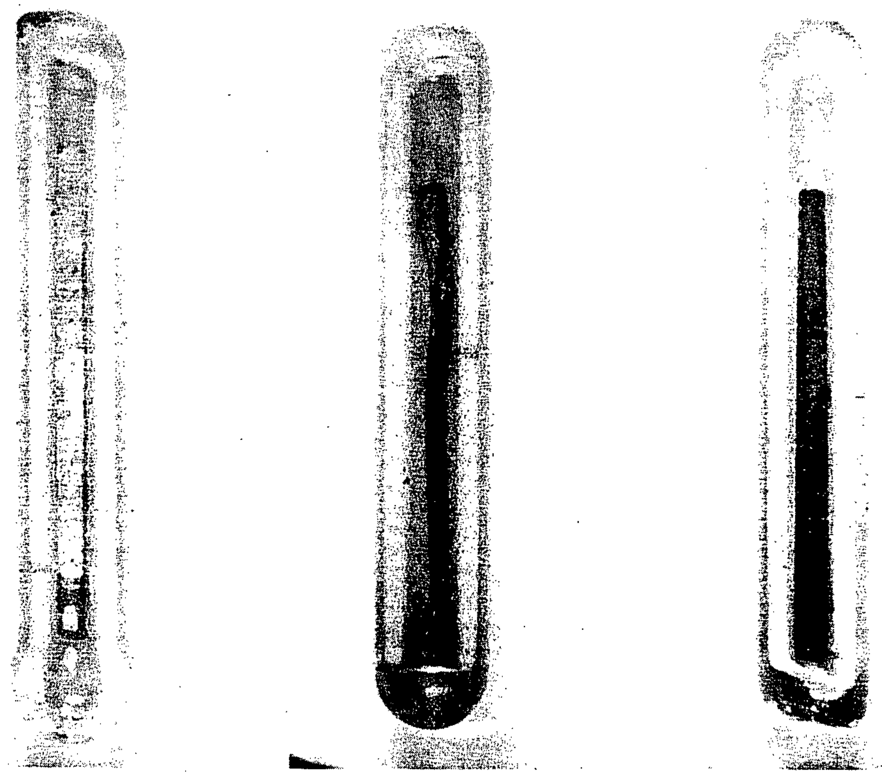
Thermal Monitor Examination

The capsule contained two 579 F (2.5 percent Ag, 97.5 percent Pb) and one 590 F (1.75 percent Ag, 0.75 percent Sn, 97.5 percent Pb) thermal monitors. The monitors were in the form of wire with a square or rectangular cross section. The 579 F monitors were located in the top and bottom regions of the capsule. The 590 F monitor was located in the middle region of the capsule.

Monitors were examined at a magnification of 5X using a stereomicroscope. Photographs are presented in Figure 2. None of the three monitors showed any evidence of melting. Based on the visual examination of the thermal monitor wires, it appears that the capsule was not above 579 F for any period of time long enough to cause melting of the thermal monitors.

Neutron Dosimetry

The surveillance capsule was in the reactor for 1237 equivalent full power days or 3.39 equivalent full power years.[16,17] The capsule was located at a core position of 55 degrees[10], and had an exposure lead



C 5083 A. Top 579 F

C 5084 B. Middle, 590 F

C 5082 C. Bottom, 579 F

FIGURE 2. PHOTOGRAPHS (5X) OF THE THREE THERMAL MONITORS FROM SURRY UNIT NO. 1 CAPSULE W

factor with respect to the inner surface of the pressure vessel wall of 0.54.[15]

Five fast neutron monitors and six thermal neutron monitors were counted for gamma ray activity to determine fast fluence (E >1 MeV) and thermal fluence, respectively. The fast flux monitors are iron, nickel, and copper, and the results are shown in Table 2. Fast fluence values (E >1 MeV) ranged from 3.5×10^{18} n/cm^2 (Fe) to 4.3×10^{18} n/cm^2 (Ni). Because the iron samples are from actual tensile specimens and the nuclear constants are well established, the average iron fast fluence (E >1 MeV) of 3.5×10^{18} n/cm^2 is considered most representative of the three monitor types. Copper values agreed within 6 percent. Nickel results should be disregarded since the half-life of the activation product is only 71.3 days, and it tends to reflect the flux level near the end of the irradiation period.

Using a lead factor of 0.54, the maximum fast fluence at the vessel wall is calculated to be 6.45×10^{18} n/cm^2 after operation for 3.39 equivalent full-power years (EFPY). The associated fast flux at the wall is 6.06×10^{10} n/cm^2/sec. Then the fluence (E >1 Mev) that would be experienced by the pressure vessel wall after 32 EFPY* is 6.12×10^{19} n/cm^2. For comparison, the value of 4.3×10^{19} n/cm^2 was originally reported in the Technical Specifications. More recently, Westinghouse stated a predicted value of 5.0×10^{19} n/cm^2.**

The experimentally determined value of 6.1×10^{19} n/cm^2 is seen to agree reasonably well (22% higher) than the Westinghouse value. Discrepancies could be caused by (1) calculation of the lead factor, (2)

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* 40 years operation at 80% load factor.

** Private communication between R. S. Denning and R. Culberson, 10/11/78.

TABLE 2. FAST NEUTRON DOSIMETRY RESULTS (E >1 MeV) FOR SURRY UNIT NO. 1

Location in Capsule	Fast Flux ($n/cm^2/sec$)			Fast Fluence (n/cm^2)		
	Iron	Nickel	Copper	Iron	Nickel	Copper
Top			3.42×10^{10}			3.66×10^{18}
Mid Top, A-6	3.32×10^{10}			3.54×10^{18}		
Middle		3.99×10^{10}			4.27×10^{18}	
Mid Bottom, V-6	3.23×10^{10}			3.45×10^{18}		
Bottom			3.49×10^{10}			3.73×10^{18}
Average	3.27×10^{10}	3.99×10^{10}	3.45×10^{10}	3.50×10^{18}	4.27×10^{18}	3.67×10^{18}

prediction of the spectrum weighted cross section, and (3) counting of the flux monitors. Of these, the largest potential source of error is in the estimation of the lead factor, which could be large enough to explain the observed variation. There also exists the possibility of a real variation of the flux at the wall as a result of a change in the power distribution in the core with time. This possibility should be discussed with the VEPCO engineers responsible for fuel management.

A final fluence value of interest is that at the 1/4 T position (1/4 of the way through the vessel wall). This has been calculated to be $0.605 \times 6.12 \times 10^{19}$ n/cm^2 = 3.7×10^{19} n/cm^2 ($E > 1$ MeV) after 32 EFPY. For further discussion see Appendix B, Neutron Dosimetry Calculations.

The thermal neutron fluences at the capsule for the bare and cadmium covered cobalt dosimeters are shown in Table 3. The true thermal fluence was equal to 2.87×10^{18} n/cm^2, calculated from $nvt_{true} = Co_{bare} \times \frac{R - 1}{R}$, where R(cadmium ratio) = $Co_{bare}/Co_{Cd\ covered} = 2.65$. Constants used in all the calculations are summarized in Table 4.

The full power flux and the actual fluence at the location of the capsule in the Surry 1 reactor were calculated from dosimeter activation analyses using the DOT 3.5 computer program. The DOT computer program yielded the neutron energy spectrum at the capsule for 22 neutron energy groups and the spectrum averaged activation cross sections for the foil flux monitors. The DECAY computer program was used to calculate the flux and fluence at the capsule. More details are given in Appendix B, Neutron Dosimetry Calculations.

TABLE 3. THERMAL NEUTRON DOSIMETRY RESULTS FOR SURRY UNIT NO. 1

Location in Capsule	Thermal Flux ($n/cm^2/sec$)		Thermal Fluence (n/cm^2)	
	Co_{bare}	$Co_{Cd\text{-}covered}$	Co_{bare}	$Co_{Cd\text{-}covered}$
Top	4.14×10^{10}	1.84×10^{10}	4.43×10^{18}	1.97×10^{18}
Middle	4.14×10^{10}	1.33×10^{10}	4.43×10^{18}	1.42×10^{18}
Bottom	4.62×10^{10}	1.69×10^{10}	4.94×10^{18}	1.80×10^{18}
Average	4.30×10^{10}	1.62×10^{10}	4.60×10^{18}	1.73×10^{18}

True Thermal Neutron Flux[(a)] = 2.68×10^{10} $n/cm^2/sec$

True Thermal Neutron Fluence = 2.87×10^{18} n/cm^2

(a) True Thermal Flux = $Co_{bare} \times \frac{R-1}{R}$

where R (Cadmium Ratio) = $Co_{bare}/Co_{Cd\text{-}covered}$ = 2.65.

TABLE 4. CONSTANTS USED IN DOSIMETRY CALCULATIONS

Reaction	Target	Isotopic Abundance (%)	Cross Section (Barns) (E >1.0 MeV)	Positional[a] Sensitivity (%/cm)	Threshold Energy (MeV)	Product Half-Life
$Fe^{54}(n,p)Mn^{54}$	96.84% Fe	5.82	0.0955	7.94	1.5	314 d
$Cu^{63}(n,\alpha)\ Co^{60}$	99.999% Cu	69.17	0.00105	4.81	5.0	5.26 y
$Ni^{58}(n,p)Co^{58}$	99.99% Ni	67.17	0.1248	5.31	1.0	71.3 d
$Co^{59}(n,\gamma)Co^{60}$	Al-0.15% Co	100	37.1 (total)	-	-	5.26 y

(a) Cross Section increases as capsule is moved toward vessel-wall.

CONCLUSIONS

Capsule W received a fast (E >1 MeV) fluence of 3.5×10^{18} n/cm^2 over the 3.39 equivalent full power years of operation. From this value it was calculated the maximum fluence seen by the inner diameter of the pressure vessel wall was 6.5×10^{18} n/cm^2. After 32 equivalent full power years of operation it is calculated that the wall inner diameter fluence and 1/4 T fluence values would be 6.1×10^{19} n/cm^2 and 3.7×10^{19} n/cm^2, respectively. Based on examination of the thermal monitors, Capsule W did not exceed 579 F.

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16. Private Communication from J. T. Benton of VEPCO to J. S. Perrin of BCL (January 31, 1975).

17. Private Communication from R. W. Calder to J. S. Perrin of BCL (September 20, 1978).

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19. RSIC Data Library Collection, CASK-40 Group Coupled Neutron and Gamma-Ray Cross Section Set, Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

APPENDIX A

LOCATION OF SURVEILLANCE CAPSULE W INSIDE SURRY UNIT NO. 1 PRESSURE VESSEL

APPENDIX A

LOCATION OF SURVEILLANCE CAPSULE E INSIDE SURRY UNIT NO. 1 PRESSURE VESSEL

The location of surveillance Capsule W inside the Surry Unit No. 1 pressure vessel is shown schematically in Figure A-1. The capsule was irradiated in the 55 degree orientation between the thermal shell and pressure vessel.

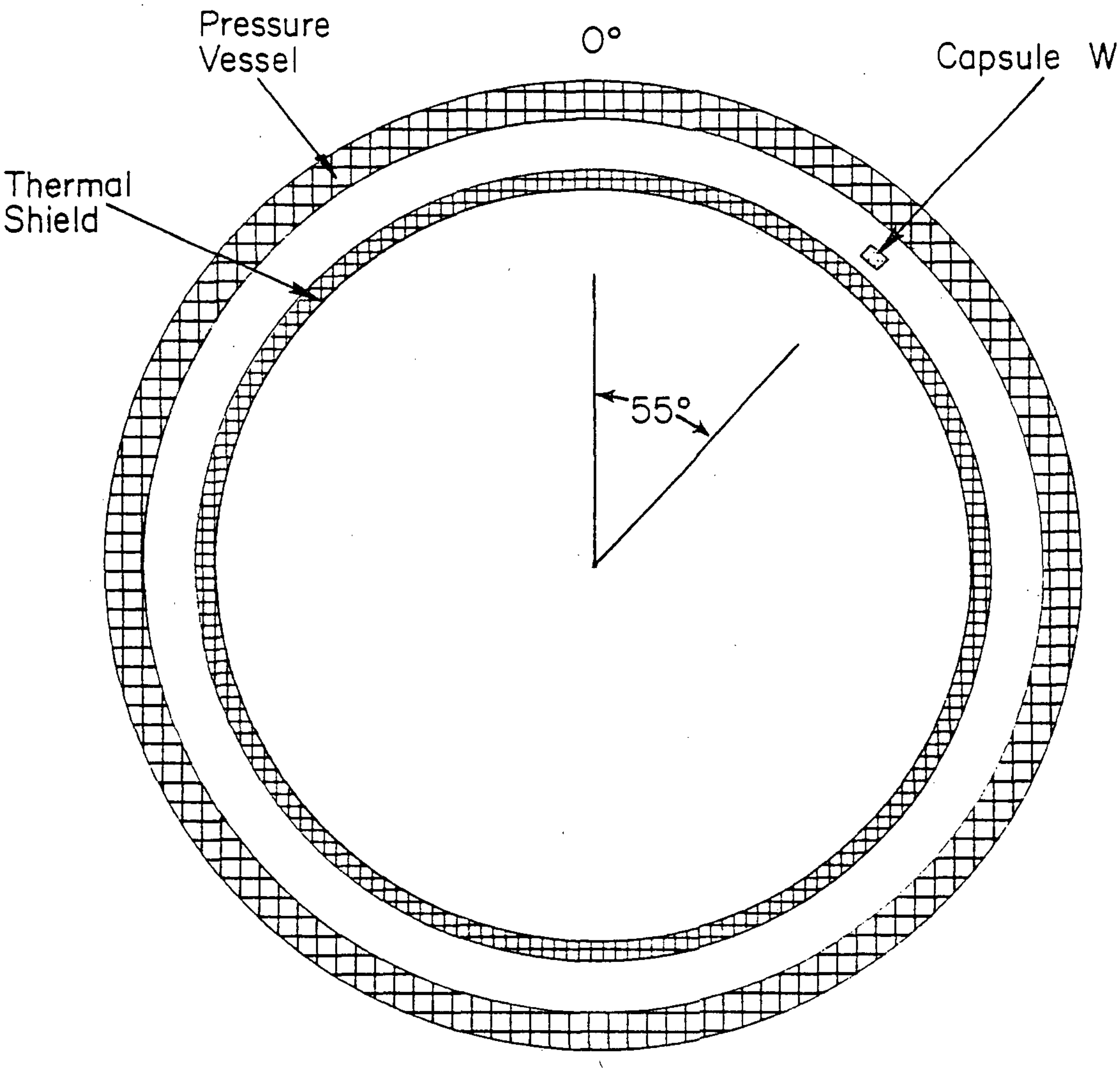

FIGURE A-1. SKETCH OF LOCATION OF CAPSULE W IN SURRY UNIT NO. 1
PRESSURE VESSEL

APPENDIX B

NEUTRON DOSIMETRY CALCULATIONS

APPENDIX B

NEUTRON DOSIMETRY CALCULATIONS

The integrated neutron fluence at a surveillance location is determined from the radioactivity induced in irradiated detector materials. A known amount of an element to be activated is placed in the neutron flux. Atoms of the dosimeter material interact with the neutron flux producing a radioactive product. After exposure, the gamma radiation from the dosimeter is measured and used to calculate the flux required to produce this level of activity. The fluence is then calculated from the integrated power output of the reactor during the exposure interval.

The activity A induced into an element irradiated for a time t_i in a constant neutron flux is given by

$$A = N \left[\int_0^\infty \sigma(E)\phi(E)dE\right] (1-e^{-\lambda t_i})$$

where

$\sigma(E)$ = the differential cross section for the activation reaction

$\phi(E)$ = the neutron differential flux

N = the atom density of the target nuclei (atoms/g)

λ = the decay constant of the product atom (sec^{-1}).

If the sample is permitted to decay for a time t_w between exposure and counting then the activity when counted is

$$A = N \left[\int_0^\infty \sigma(E)\phi(E)dE\right](1-e^{-\lambda t_i})\, e^{-\lambda t_w} \quad .$$

A "spectrum-averaged cross section" may be defined as

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

and the integrated flux as

$$\phi = \int_0^\infty \phi(E)dE \quad .$$

Then

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

so that the activity A may be written as

$$A = N\sigma\phi(1-e^{-\lambda t_i})\, e^{-\lambda t_w} \quad .$$

The flux is then computed from the measured activity as

$$\phi = \frac{A}{N\sigma(1-e^{-\lambda t_i})\, e^{-\lambda t_w}} \quad .$$

If it is desired to find the flux of neutrons with energies above a given energy level E_c, the cross section corresponding to this energy level is defined as

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

where

$$\phi(E>E_c) = \int_{E_c}^{\infty} \phi(E)dE \quad .$$

Then

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

$$= \sigma(E>E_c)\ \phi(E>E_c)$$

and the activity A may be written as

$$A = N\sigma(E>E_c)\phi(E>E_c)(1-e^{-\lambda t_i})e^{-\lambda t_w} \quad .$$

In case that the neutron flux is not constant the dosimeter activity at the time of removal from the reactor is

$$A = N\sigma(E>E_c)\phi(E>E_c)\ C$$

where

$$C = \sum_{j=1}^{J} f_j\ (1-e^{-\lambda T_j})e^{-\lambda(T-t_j)}$$

J = number of time intervals of constant flux

f_j = the fractional power level during the time interval J

T_j = the time length of interval j

t_j = the elapsed time from beginning of irradiation to end of interval j

T = the time from beginning of irradiation to counting.

Then

$$\phi(E>E_c) = \frac{A}{N\sigma(E>E_c)C} .$$

This is the equation used to find fluxes based on surveillance dosimeter activations. The time intervals are taken as one month each and average power during the month is used for f values. A DOT 3.5[18] calculation was performed to find the spectrum averaged activation cross sections for the flux monitors. DOT is a computer program which solves the Boltzmann transport equation in two-dimensional geometry. The method of discrete ordinates is used. Balance equations are solved for the density of particles moving along discrete directions in each cell of a two-dimensional spatial mesh. Anisotropic scattering is treated using a Legendre expansion of arbitrary order.

The two-dimensional geometry that was used to model the Surry reactor is shown in Figure B-1. As seen, there are 7 circumferential divisions and 58 radial divisions. Capsule W was taken to be a 1-inch square of steel and includes circumferential meshes 6 and 7 and radial meshes 45, 46, and 47. Third order scattering was used (P_3) and 48 angular directions of neutron travel (24 positive and 24 negative) (S_8) were used. Neutron energies were divided into 22 groups ranging from 14.9 MeV energy down to 0.01 MeV energy. The 22 group neutron structure is that of the RSIC Data Library DLC/Cask[19], and neutron absorption, scattering, and fission cross sections used are those supplied by this library. The baffle, barrel, and thermal shield are stainless steel type 304. The reactor vessel is A533 Grade B Class I material. The reactor core was mocked up as homogenized

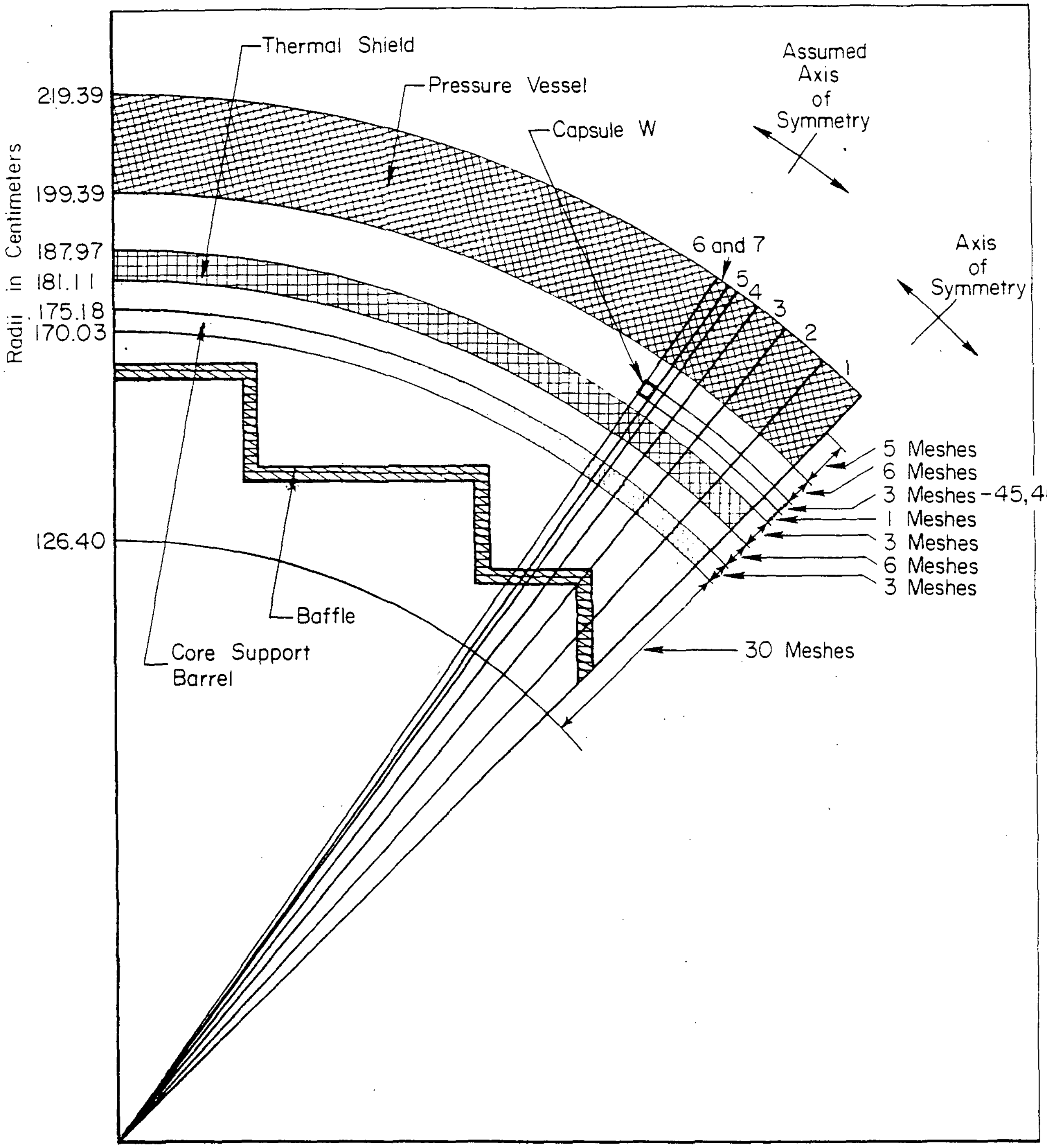

FIGURE B-1. SURRY GEOMETRY USED IN DOT COMPUTER RUN fuel and water having the same densities as found in the operating reactor. The fuel was a source of neutrons having a U^{235} fission energy spectrum.

The neutron spectrum at the center of the capsule, as calculated by DOT, is shown in Figure B-2. Also shown for comparison is the fission spectrum. Both spectra have been normalized to contain one neutron above 1.0 MeV. It is seen that the DOT spectrum and the fission spectrum do not differ appreciably. However, the DOT calculated values of "spectrum averaged" cross section σ_R differ from the fission-spectrum-averaged cross sections by as much as 50 percent (for copper) and by about 20 percent for iron.

Table 4, in the main body of the text, also gives the constants used in calculating the neutron flux. These, together with the reactor power history were used in the DECAY computer program to calculate the flux and fluence at the capsule. The power history was taken from two sources. The first is a private communication from J. T. Benton to J. S. Perrin,[16] which supplied the power history from start-up through October 21, 1975, when Capsule T was removed. The second is a private communication from R. W. Calder to J. S. Perrin,[17] which supplied the operating history from start-up on February 1, 1975, through shut-down on April 22, 1978, when Capsule W was removed. The equivalent full power days of operation until removal of Capsule T is 391.0 and until removal of Capsule W is 1237.16.

The fast flux distribution in the vessel wall as calculated by ANISN is

Surface	Fast Flux = 1.000
1/4 through wall	Fast Flux = 0.605
3/4 through wall	Fast Flux = 0.155

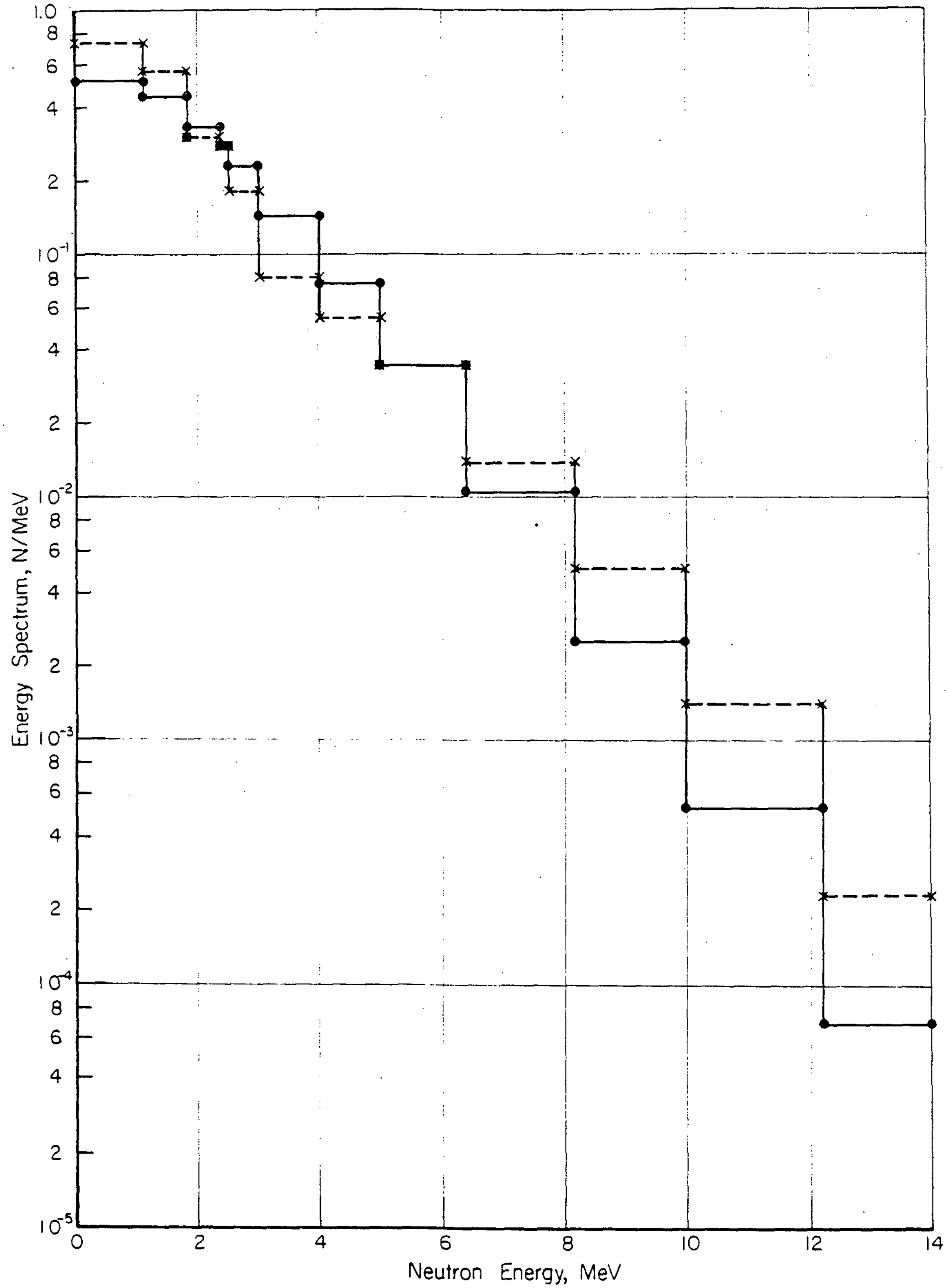

FIGURE B-2. COMPARISON OF FAST NEUTRON SPECTRUM AT THE CAPSULE WITH FISSION SPECTRUM

These are calculated to exist in the vessel wall immediately behind Capsule W, and it is believed they would be representative of the distribution through the vessel wall at any radial position about the core.