CALCULATION OF GAMMA FLUXES IN CANDU LATTICES

C.R. BOSS

Atomic Energy of Canada Limited
Engineering Company
Sheridan Park Research Community
Mississauga, Ontario L5K 1B2

ABSTRACT

The discrete ordinates codes ANISN and DOT have been used to calculate gamma fluxes in the lattice of a CANDU reactor. The libraries of cross sections used and details of the calculations are described.

The gamma fluxes are used to calculate where gamma energy is deposited in the lattice, photoneutron sources in the shutdown reactor and the gamma response of the platinum self-powered detectors. Comparisons with measured values show reasonable agreement.

INTRODUCTION

In the reactor the main sources of gamma radiation are:

1) the prompt gammas released in fission
2) the decay gammas released by the fission products

and 3) the prompt gammas released in neutron capture.

These gamma sources represent about 10% of the nuclear energy released in the operating core. At shutdown the decay gammas from the fission products are the main source of gamma radiation and they then represent about 50% of the nuclear energy released.

In both cases we need to know where the gamma energy appears as heat. For that reason we need to calculate the gamma fluxes in the components of the lattice cell.

The gamma fluxes also contribute to the signal from the platinum self-powered flux detectors used to monitor the power of the core. Changes in the gamma fluxes due to changes in the gamma sources, e.g., during refuelling, will lead to changes in the signals from the detectors. To relate the changes in detector signal to changes in reactor power we need to calculate the changes in gamma fluxes at the detectors.

CALCULATIONS

a) Gamma Fluxes in Lattice Components of an Operating Core

Our study of gamma fluxes in the lattice components of an operating core used the one-dimensional discrete ordinates code ANISN (1).

In this study the 28.6 cm square lattice of the 37-element CANDU core was represented by twelve regions in cylindrical geometry as shown in Figure 1. In this model the thirty-seven elements were represented by a central fuel element and three fuel annuli whose areas equalled the cross sectional area of the fuel in the inner, intermediate and outer rings of the 37-element bundle. The fuel element void, the Zircaloy cladding and the heavy water coolant were homogenized in four "coolant + cladding" annuli. At the outside of the moderator region a reflecting boundary condition was used to simulate the repeating lattice of the reactor.

The ANISN calculation was run in the fixed source mode using the library of coupled-neutron gamma cross sections described in the Appendix. The fixed source had a fission neutron spectrum. No fixed gamma source was input because the ANISN calculation automatically created the prompt fission, fission product decay and neutron capture gamma sources. Thus fixed source terms were input for the top fourteen groups of the thirty-nine groups of Table 1 with relative source strengths in the four fuel regions taken from a WIMS (2) calculation.

b) Gamma Fluxes in Lattice Components of a Shutdown Core

Our study of gamma fluxes in the lattice of a shutdown core was in two steps. For the first step we used the code ORIGEN (3) to find the fission product inventories and associated gamma source spectra at different times after shutdown. For the second step we used a series of ANISN calculation to find the gamma fluxes arising from these ORIGEN source spectra.

The ANISN calculations were run in the fixed source mode using the ORIGEN library of cross sections described in the Appendix. The geometry used for these calculations was the same as that used for the operating core. The spatial distribution of the gamma source in the four fuel regions followed the fission source distribution of the WIMS calculation.

In the first ANISN run we calculated the gamma fluxes generated in the lattice by a gamma released in the top energy group of Table 3. In eleven other ANISN runs we calculated the gamma fluxes generated by a gamma released in each one of the lower energy groups.

The total gamma flux distributions were then calculated by summing the gamma flux distributions from the twelve ANISN calculations weighted according to the ORIGEN source spectrum. In this way we avoided running ANISN calculations for every ORIGEN source spectrum.

c) Gamma Fluxes in Lattice During Refuelling

Our study of changes in the gamma fluxes during refuelling used the two dimensional discrete ordinates code DOT 3.5 (4).

In this study 3-1/2 x 3 lattice pitches of the 28.6 cm square lattice were represented in XY geometry as shown in Figure 2. For each channel the thirty-seven fuel elements, the Zircaloy cladding and the heavy water coolant were homogenized into one region. The pressure tube, the gas annulus and the calandria tube were also homogenized.

Two calculations were run for this geometry in the fixed source mode. The calculations used the fission neutron source distributions obtained from two three-dimensional diffusion calculations; the first had all the channels fuelled, the second had an empty channel. Again we used the coupled neutron-gamma library; no gamma source was input.

RESULTS

a) Heating in Lattice Components Of an Operating Core

Typical values of the gamma fluxes in the lattice of an operating core calculated by ANISN are shown in Table 4. However, our intention was to calculate $Q(r)$, the rate at which gamma energy is deposited as a function of position in the lattice cell where

$$Q(r) = \int \phi(E,r)\mu_e(E,r)EdE$$

and $\phi(E,r)$ is the gamma flux as a function of energy E and position r while $\mu_e(E,r)$ is the energy absorption cross section as a function of energy and position.

In the multigroup formalism of the discrete ordinates calculation this integral becomes a summation over j groups thus:

$$Q(r) = \sum_j \phi_j(r)\mu_{ej}(r)E_j$$

and the total gamma energy deposited in one of the twelve regions of the ANSIN calculation

$$H_k = 2\pi\int_{r_{in}}^{r_{out}} Q(r)\,r\,dr$$

is given by

$$H_k = \sum_j \Phi_{jk}\mu_{ej}E_j$$

where Φ is the region integrated flux.

We have used the activation option of ANISN to calculate these integral quantities. The results for an operating core are presented in Table 5.

The distribution of heat produced as the fission neutron slowed down in the lattice cell was also calculated by ANISN.

The total of the neutron and gamma energy deposited in the moderator, calandria tubes and in-core structures appears as the heat load across the moderator heat exchanger. Our estimate of the moderator heat load for Bruce NGS "A" derived from the ANISN fluxes is within 5% of the measured value. This is well within the $\pm$ 11% uncertainty of the measured value.

These calculations showed that the heat removed by the primary coolant as a fraction of the total fission energy released is 0.7% higher than the fraction derived with earlier less sophisticated calculations. That is, the reactor is slightly more efficient than had been calculated previously.

b) <u>Signal from the Platinum Self-Powered Flux Detectors</u>

i) <u>Effect of an Empty Channel</u>

About 60% of the signal from the platinum clad self-powered detectors used to monitor reactor power is due to thermal neutron interactions in the detector. The remaining 40% of the signal is due to the external gamma interactions. Thus, to find the change in detector signal when a fuel channel is emptied, we need to calculate the changes in thermal neutron and gamma flux at the detector.

Changes in thermal neutron fluxes were obtained from reactor physics calculations which calculated the thermal neutron flux and power distributions in the core before and after the channel is emptied.

Changes in gamma flux were obtained from the DOT 3.5 calculations described above. To assess the overall effect of the changes in each gamma group we assumed that the gamma response of the detector was proportional to the platinum total gamma absorption cross section.

i.e. $$R \propto \int_E \phi(E)\sigma_t(E)\,dE$$

where R is the overall gamma response of the detector

where ϕ is the gamma flux at the detector as a function of energy E.

where σ_t is the total absorption cross section of the detector as a function of energy E, which in a multigroup formalism becomes

$$R \propto \sum_j \phi_j \sigma_{tj}$$

where j represents the energy group.

The results of both the reactor physics and DOT 3.5 calculation are shown in Figure 3 for the detectors which are one lattice pitch long.

For the detector nearest the defuelled channel the gamma component of the detector signal decreased by 14.1% while the neutron component increased by 1.1%. The net effect was

$$(0.6 \times 1.1) - (0.4 \times 14.1) = -5.0\%$$

that is a 5% decrease in signal. The average decrease in power in the surrounding fuel is only 1.3%. Further away from the defuelled channel, the gamma flux changes are nearly proportional to the neutron flux changes; hence the detector signal changes closely represent the power changes in the local fuel.

ii) Effect During Shutdown

After shutdown the neutron flux usually drops much lower than the gamma flux. Because of the gamma sensitivity of the platinum detectors, the detector signals after shutdown will be proportional to gamma flux rather than neutron flux.

From a series of measurements in various reactors C.J. Allen et al have developed a set of decay constants which represent the decay of the signal from a platinum clad detector in a CANDU lattice. A "measured" response has been constructed in Figure 4 using Allen's decay components as shown by the measurement points in the figure. The expected decay signal after a 95 day irradiated was calculated with ORIGEN and ANISN and is shown for comparison with the "measured" points. The calculated curve is normalized to the measured points at 0.1 hours after shutdown. This shows a fairly good correlation between measurement and the ORIGEN/ANISN calculation.

c) Photoneutron Sources in Shutdown Core

The gamma fluxes from the twelve ANISN calculations for the shutdown core can also be used to find the photoneutron sources at shutdown.

In the shutdown reactor the fission products provide a source of gammas with energy above the 2.23 MeV threshold of the photoneutron reaction $^2H(\gamma,n)^1H$. Thus the primary coolant and moderator become neutron sources. These neutron sources keep the reactor at very low power levels which can be measured by the ion chambers.

We have therefore used the gamma fluxes from the ANISN calculations for the shutdown core to find the photoneutron sources,

$$S = 2\pi \int r\,dr \int_{E > 2.23\text{MeV}} \phi(r,E)\sigma(r,E)\,dE$$

in both the moderator and coolant regions.

We have used the cross section data given in Reference 5 and evaluated the integral for the moderator and coolant regions. This procedure was followed for only seven of the ANISN calculations since a source gamma emitted in group 8 is below the threshold of the photodisintegration of deuterium. In this way we calculated the probability that a gamma released in one of the seven groups will generate a photoneutron.

The probabilities are given in Table 6. To find the generation rate of photoneutron sources we sum the probabilities weighted by the absolute source spectrum from our ORIGEN calculations.

We calculated the photoneutron generation rate in two units of the Bruce NGS "A" reactor following irradiations of several months and in a third unit following an irradiation of five hours. In each case we compared the measured and calculated ion chamber signals as a fraction of the full power signal.

After the irradiations of several months the reactor was shutdown by adding poison. Since this caused a relatively small flux redistribution we treated the core as a point reactor and assumed that the ion chamber signal was proportional to the rate at which neutrons were generated. In the shutdown core the neutron generation rate was simply the product of the photoneutron generation rate and the subcritical multiplication factor of the shutdown core, $1/(1-k_{eff})$. We also applied a correction for the effect of the poison in the reflector on the ion chamber signal. The results are shown in Figures 5 and 6. Between 11 and 15 days after shutdown the agreement shown in Figure 5 is fairly good. The reason for the discrepancies at earlier and later times has not been investigated.

In the case of the five hour irradiation the reactor was shutdown by shutoff rod insertion. This caused a large redistribution in the neutron flux thus changing the relationship between the ion chamber signal and the photoneutron source. In this case the decay transient was calculated with a three dimensional time dependent diffusion code with the delayed photoneutron parameters obtained from our ORIGEN and ANISN calculations.

The comparison with measurement is shown in Figure 7 for two ion chamber locations.

CONCLUSIONS

We have used the discrete ordinates codes ANISN and DOT 3.5 and the fission product inventory code ORIGEN to find the gamma fluxes in the lattice of a CANDU reactor. From these fluxes we have calculated three integral quantities, namely, nuclear heating in the lattice, the gamma response of the platinum detectors and the photoneutron sources in the shutdown core.

The agreement between calculation and measurement shows that this approach is a reasonably accurate method of calculating the integral quantities.

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APPENDIX

Library of Coupled Neutron-Gamma Cross Sections

The coupled neutron-gamma library used in our operating reactor calculations has twenty-seven neutron groups and twelve gamma groups. The group structure of the library is shown in Table 1.

The group structure was based on our experience with earlier coupled neutron-gamma libraries. That experience showed we needed a gamma group with mean energy 0.51 MeV to handle the annihilation gammas created via pair production. We also found that our calculations of gamma energy deposition would be improved if we had more than one group below 0.5 MeV. Finally we found it convenient to include groups with mean energy 1.25 MeV (approximately energy of ^{60}Co decay gamma and 6.0 MeV approximately the energy of ^{16}N decay gammas).

We could continue to improve the resolution of the gamma flux spectra with finer group structures. Unfortunately the cost of a discrete ordinates calculation is proportional to the number of groups. Our final choice of the 12-group structure is therefore a compromise between the resolution of the gamma spectra and cost. Our comparisons with measurement suggest that the 12-group structure is an acceptable compromise.

The procedure used to create the coupled neutron-gamma library is shown in Figure 1.

In the first step the 100-group neutron cross sections of the DLC-2 library (6) were used in an ANISN calculation of neutron fluxes in the lattice cell. The 100-group neutron fluxes were used to weight the 100-group cross sections to produce 27-group neutron cross sections.

The second step is more involved. We need a matrix of neutron to gamma cross sections which will generate the gamma sources from,

fission
neutron capture
fission product decay
inelastic neutron scatter
beta decay of activation products
n, 2n scatter

In the case of delayed emission, such as beta decay, equilibrium was assumed so that the process could be treated as an instantaneous one in the same way as, say, neutron capture gamma sources are expressed.

The sources of gammas of energy E produced at r in one of these processes is,

$$S(r,E) = \int_{E_n} \phi(r,E_n)\Sigma(r,E_n)Y(E_n,E)\,dE_n$$

where $\phi(r,E_n)$ is the flux of neutrons of energy E_n at r

where $\Sigma(r,E_n)$ is the cross section of the process at neutron energy E_n

and $Y(E_n,E)$ is the yield of gammas of energy E per interaction.

In multigroup formalism this is,

$$S_j(r) = \sum_i \phi_i(r)\Sigma_k(r)Y_{ij}$$

where i is the neutron energy group and j is the gamma energy group and the product $\Sigma_i(r)\ Y_{ij}$ is the neutron to gamma cross section, i.e., the probability of a neutron being scattered from neutron group i to release Y_{ij} gammas in group j.

Details of the gamma yields were taken from the DLC-12 library (7) and the neutron cross sections for fission and neutron capture were taken from the 27-group neutron cross sections. The neutron cross sections for inelastic and (n,2n) scatter were obtained from Reference 8.

The POPOP4 (9) code processes these yield and cross section data for any number of interactions to derive a single matrix of neutron to gamma cross sections.

Note that the data for xenon took account of the gammas released by neutron capture in all the fission products present in a CANDU reactor which has reached fuelling equilibrium.

Also, to study the variation in the fission product gamma source during the first few hours of irradiation of a core of fresh fuel, as well as for a core at equilibrium, four separate matrices of neutron to gamma cross sections were set up for, ^{235}U and ^{238}U, for the following irradiations:

zero
3×10^3 seconds
2×10^4 seconds
equilibrium

For ^{239} Pu only the equilibrium irradiation matrix was set up.

In the third step we used the program MUG (10) to calculate a matrix of cross sections for the gamma interactions;

pair production
Compton scatter
photoelectric effect

The program calculates the Compton scatter cross section from an analytic expression and reads the cross sections for pair production and the photoelectric effect from the DLC-7 library (11).

Finally the three matrices of neutron, neutron to gamma and gamma cross sections are combined using the sample simple coupling code SSCC (12).

This procedure was followed for the nineteen elements and isotopes listed in Table 2.

Library of ORIGEN Gamma Cross Sections

For our calculations in the shutdown core we have found it convenient to derive a set of gamma cross sections in the 19-group structure of ORIGEN (see Table 3). The library contains data for hydrogen, carbon, oxygen, zirconium and uranium.

To create this library of cross sections we ran the MUG program, i.e., the third step of the procedure in Figure 8.

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TABLE 1

ENERGY STRUCTURE OF THE 39-GROUP COUPLED NEUTRON-GAMMA LIBRARY

Group	Upper Energy Limit (eV)	Group	Upper Energy Limit (eV)
Neutrons		Gammas	
1	1.49(7)*	28	1.00(7)
2	1.22(7)	29	7.00(6)
3	1.11(7)	30	5.00(6)
4	6.06(6)	31	4.00(6)
5	3.68(6)	32	3.00(6)
6	2.23(6)	33	2.00(6)
7	1.35(6)	34	1.50(6)
8	8.21(5)	35	1.00(6)
9	4.98(5)	36	0.62(6)
10	3.02(5)	37	0.40(6)
11	1.83(5)	38	0.20(6)
12	6.74(4)	39	0.12(6)
13	4.09(4)		
14	2.48(4)		
15	1.50(4)		
16	5.53(3)		
17	2.03(3)		
18	7.49(2)		
19	2.75(2)		
20	1.01(2)		
21	4.79(1)		
22	2.26(1)		
23	1.07(1)		
24	3.93(0)		
25	1.45(0)		
26	6.83(-1)		
27	4.14(-1)		

* Read 1.49(7) as 1.49×10^7

TABLE 2

MATERIALS IN THE 39-GROUP COUPLED NEUTRON-GAMMA LIBRARY

hydrogen	iron
deuterium	nickel
boron-10	zirconium
boron-11	xenon
carbon	gadolinium
oxygen	lead
aluminium	uranium-235
silicon	uranium-238
chromium	plutonium-239
manganese	

TABLE 3

GAMMA ENERGY GROUPS USED IN ORIGEN AND ORIGEN LIBRARY OF GAMMA CROSS SECTIONS

Group	Lower Energy Bound MeV	Mean Energy MeV
1	5.0	5.25
2	4.5	4.70
3	4.0	4.22
4	3.5	3.7
5	3.0	3.2
6	2.6	2.75
7	2.2	2.38
8	1.8	1.99
9	1.35	1.55
10	0.9	1.10
11	0.4	0.63
12	0.25	0.3
13	0.175	0.2
14	0.125	0.15
15	0.075	0.1
16	0.050	0.06
17	0.035	0.04
18	0.020	0.03
19	0.010	0.015

TABLE 4

GAMMA FLUXES IN 37-ELEMENT LATTICE OF OPERATING CORE

Group	Gamma Fluxes ($\gamma/cm^2 \cdot s$) per Fission Neutron at Interfaces between		
	Coolant/Pressure Tube	Calandria Tube/Moderator	at Cell Boundary
28	6.05(-5)	5.12(-5)	2.53(-5)
29	1.29(-3)	1.14(-3)	6.31(-4)
30	2.82(-3)	1.99(-3)	8.37(-4)
31	7.13(-3)	4.93(-3)	2.02(-3)
32	2.26(-2)	1.52(-2)	6.21(-3)
33	2.05(-2)	1.35(-2)	5.29(-3)
34	3.10(-2)	2.07(-2)	8.36(-3)
35	3.26(-2)	2.27(-2)	1.06(-2)
36	3.00(-2)	2.22(-2)	1.26(-2)
37	2.60(-2)	2.88(-2)	2.73(-2)
38	6.49(-3)	1.67(-2)	2.47(-2)
39	4.56(-5)	2.91(-3)	7.92(-3)

TABLE 5

GAMMA HEAT DISTRIBUTION IN 37-ELEMENT LATTICE COMPONENTS

Region	Gamma Energy Deposited MeV per Fission Neutron	Fraction of Total Gamma Energy Deposited in Lattice Components
Central Pin	0.190	0.019
Coolant	0.029	0.003
Cladding	0.044	0.004
Inner Fuel Ring	1.150	0.115
Coolant	0.054	0.005
Cladding	0.080	0.008
Intermediate Fuel Ring	2.319	0.232
Coolant	0.088	0.009
Cladding	0.132	0.013
Outer Fuel Ring	3.312	0.331
Coolant	0.028	0.003
Cladding	0.044	0.004
Pressure Tube	0.510	0.051
Gas Annulus	-	-
Calandria Tube	0.178	0.018
Moderator	1.847	0.185

TABLE 6

PROBABILITY OF PHOTONEUTRON PRODUCTION

ORIGEN Group Lower Energy (MeV)	Probability of Photoneutron Production Per Unit Gamma Source
5.0	1.44 (-3)
4.5	1.39 (-3)
4.0	1.32 (-3)
3.5	1.17 (-3)
3.0	9.59 (-4)
2.6	6.67 (-4)
2.2	2.34 (-4)

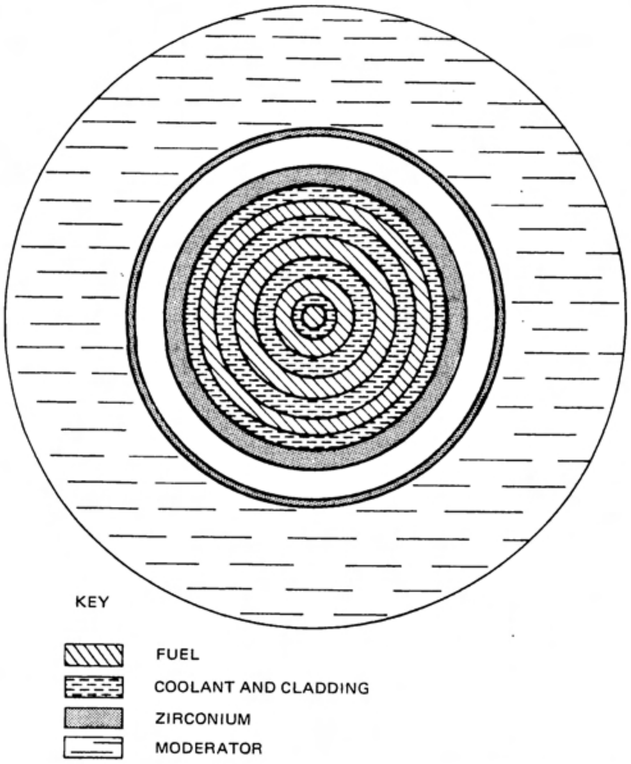

FIGURE 1 GEOMETRY USED TO REPRESENT THE 37-ELEMENT FUELLED LATTICE IN ANISN CALCULATIONS

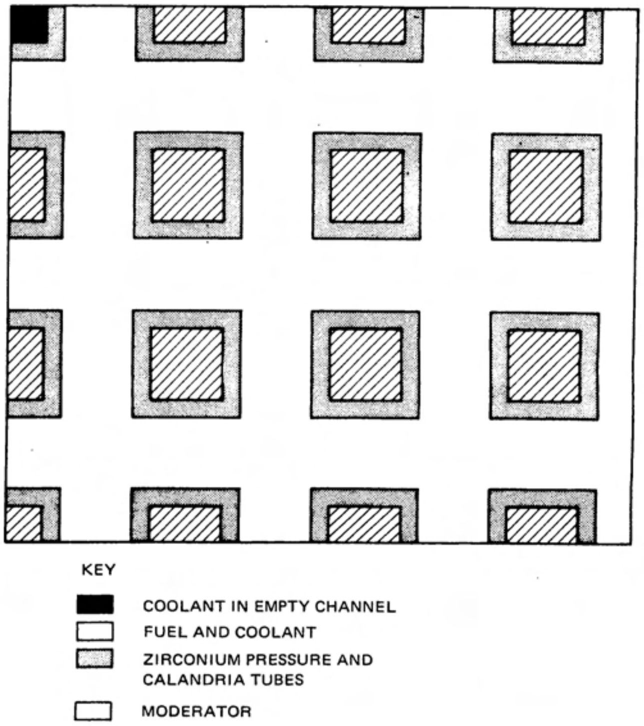

FIGURE 2 GEOMETRY OF TWO DIMENSIONAL DOT 3.5 CALCULATIONS USED TO STUDY CHANGES IN GAMMA FLUXES DURING REFUELLING

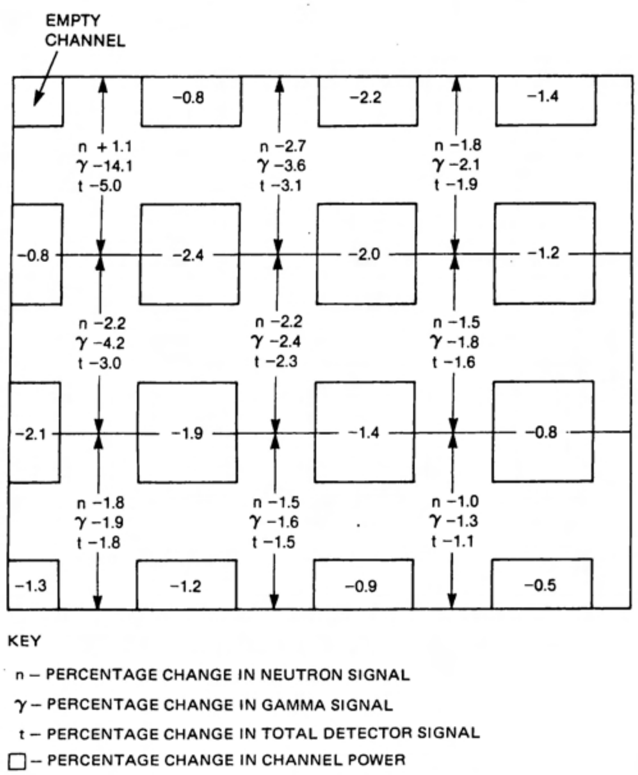

FIGURE 3 EFFECT OF AN EMPTY CHANNEL ON CHANNEL POWER AND THE SIGNAL FROM ONE LATTICE PITCH LONG DETECTORS LOCATED IN AXIAL MID-PLANE

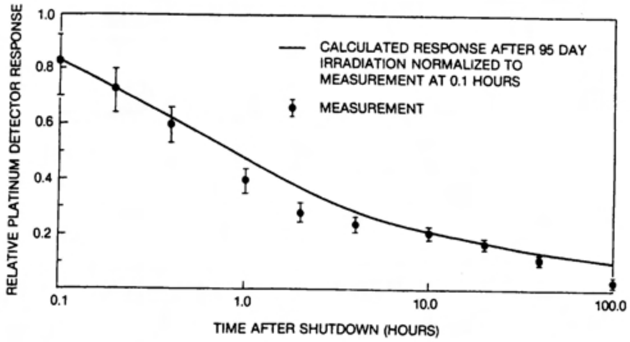

FIGURE 4 RELATIVE RESPONSE OF PLATINUM DETECTOR AFTER A SHUTDOWN - COMPARISON BETWEEN CALCULATION AND MEASUREMENT

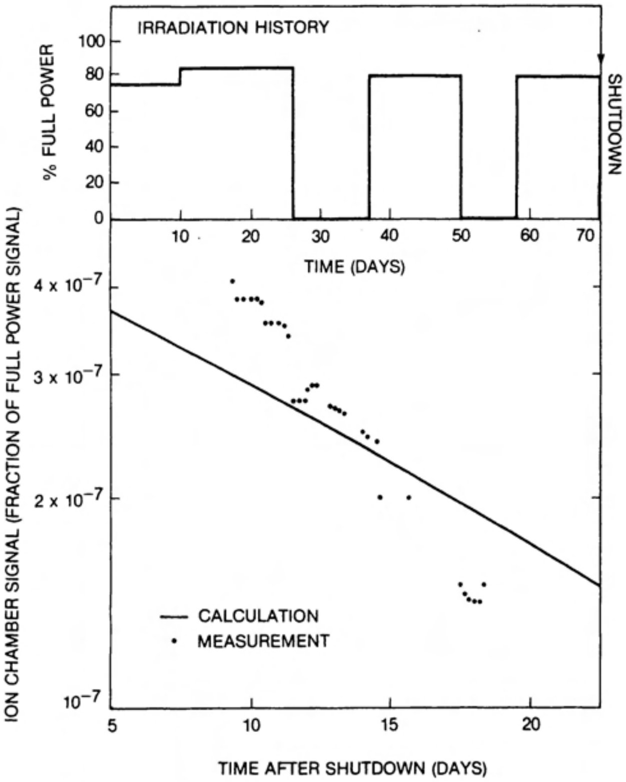

FIGURE 5 COMPARISON OF THE CALCULATED ION CHAMBER SIGNAL WITH MEASUREMENT

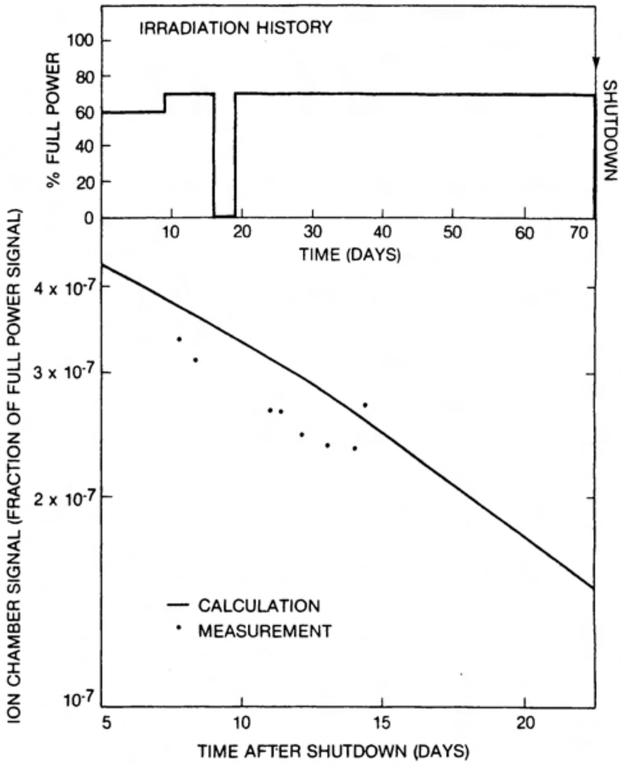

FIGURE 6 COMPARISON OF THE CALCULATED ION CHAMBER SIGNAL WITH MEASUREMENT

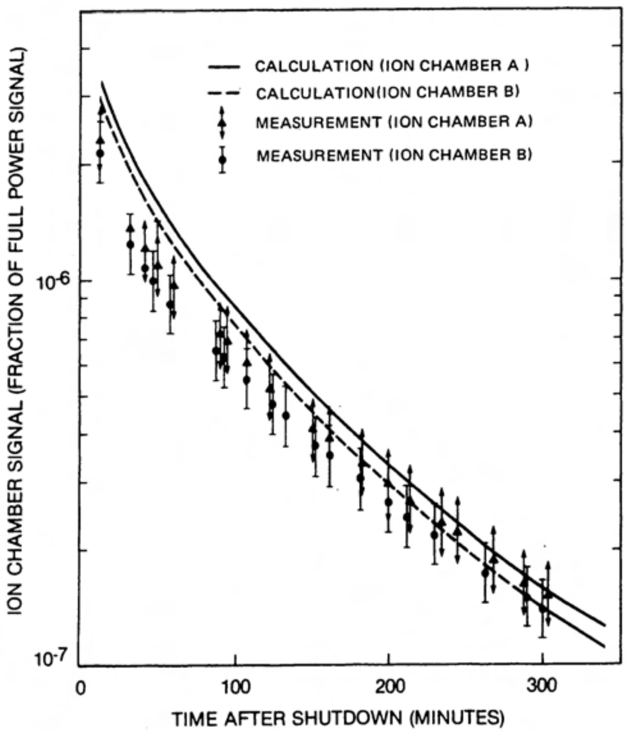

FIGURE 7 COMPARISON OF THE CALCULATED ION CHAMBER SIGNAL WITH MEASUREMENT AFTER A FIVE-HOUR IRRADIATION

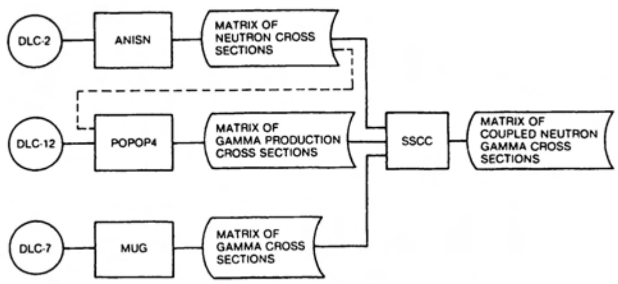

FIGURE 8 PROCEDURE USED TO CREATE THE 39-GROUP COUPLED NEUTRON-GAMMA LIBRARY