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ENVIRONMENTS OF C AND K REACTOR SPECIALTY
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G. D. Guthrie

ENVIRONMENTS OF C AND K REACTOR SPECIALTY
IRRADIATION FACILITIES

Purpose

This document describes the neutron and gamma environments in Hanford Specialty Irradiation Facilities, and discusses the effects of reactor operation on these environments.

Neutron Fluxes

The neutron fluxes described below are calculated fluxes, partially supported by experimental data. They should be interpreted as,

$$\phi_i = \int_{v_1}^{v_2} N(v) v dv$$

for the energy ranges indicated. Throughout this document, the energy group structures are:

<u>Group</u>	<u>Energy Range</u>
1	$E > 1.05 \text{ Mev}$
2	$5.53 \text{ kev} < E \leq 1.05 \text{ Mev}$
3	$0.415 \text{ ev} < E \leq 5.53 \text{ kev}$
4	$E \leq 0.415 \text{ ev}$

The neutron fluxes in Table I are averaged over a cell boundary, where the cell is a member of an infinite lattice of identical cells, with the fuel operating at 100 kw/ft. The K-Reactor lattice cells are described in Appendix I.

Table I: Fluxes averaged over a cell boundary.

<u>Group</u>	<u>Natural Fueled Cell (neut/cm²-sec)</u>	<u>.947% Enriched Fueled cell (neut/cm²-sec)</u>
1	1.0×10^{13}	1.0×10^{13}
2	5.0×10^{13}	5.0×10^{13}
3	7.5×10^{13}	7.2×10^{13}
4	1.7×10^{14}	1.5×10^{14}

The neutron fluxes in Table II are for a point on the inner surface of a front-to-rear facility. The facility is nearly full of aluminum (the details of the cell are shown in Figure I), and is surrounded by either natural or .947% enriched cells. In both cases, the surrounding fuel is operating at 100 kw/ft.

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Table II: Neutron flux at inner surface of front-to-rear facility.

Group	Natural Fueled Cells (neut/cm ² -sec)	.947% Enriched Fueled Cells (neut/cm ² -sec)
1	5.3×10^{12}	5.4×10^{12}
2	3.4×10^{13}	3.4×10^{13}
3	5.1×10^{13}	4.9×10^{13}
4	1.7×10^{14}	1.5×10^{14}

Homogenized cross sections, for natural and enriched cells, are listed in Appendix II.

Total Heating In Nonfissionable Specimens

The gamma spectrum in Hanford Reactors is not well defined. The method described below for calculating total heat generation in nonfissionable specimens is taken from reference 1, and predicts higher than actual total heating rates for materials having (n,γ) reactions. Materials, such as B¹⁰, which have (n,α) reactions, may have heat generation rates orders of magnitude higher.

The three significant heat generating processes are:

1. The absorption in the specimen of fission gammas and gammas produced by fission product decay.
2. The decay of radioactive isotopes produced in the specimen.
3. The absorption in the specimen of secondary gammas (those produced by an n,γ reaction).

The first process is proportional to the reactor power and is a function of the specimen material. Based on previous tests, a constant of 0.6 watts per gram is suggested for this contribution.

The secondary process varies considerably for different elements but is usually negligible. One should be wary of isotopes decaying with alpha emission, in which case, nearly all of the alphas will be absorbed in the specimen.

Most (n,γ) reactions release about 7.5 Mev per neutron absorbed. This energy is manifest in characteristic "capture gamma" photons which generate heat through attenuation in lattice materials. If we assume the constant 7.5 Mev per neutron absorbed, then we can determine the heating contribution of process 3 by determining neutron absorption rates, and predicting the portion of the secondary gammas that will be attenuated in the specimen.

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$$NE \text{ (Total Energy Released by Neutron Capture)} = E_b f_{ss} \phi_0 \Sigma_a V$$

where E_b = binding energy per nucleon (≈ 7.5 Mev)
 f_{ss} = self-shielding factor (dependent on geometry and absorption cross section)
 ϕ_0 = unperturbed neutron flux
 V = volume of the target specimen
 Σ_a = macroscopic absorption cross section of the specimen

The self-shielding factor, f_{ss} , is defined as $\bar{\phi}/\phi_0$, where $\bar{\phi}$ is the average flux in the sample. The self-shielding factor is usually presented as a function of the parameter T , where $T = 2V\Sigma_a/S$, and S is the surface area of the specimen. For most solid bodies, if $T \geq 2$, then $f_{ss} \approx T/2$, and,

$$NE = \frac{7.5 \text{ Mev } \phi_0 S}{4}$$

The curves of Figure II may be used to determine f_{ss} for other values of T .

We now have the energy represented in "capture gamma" photons and must determine what portion of these gammas will be stopped in the specimen. Curves of Figure III approximate the gamma attenuation factor (f_{ga}) for a solid rod. The total heat generation in a specimen is, then,

$$E_T = (NE \times f_{ga}) + 0.6 \text{ watts/gm}$$

An in-reactor calorimeter has been used to measure total heat generation in representative nonfissionable specimens. These data are now being analyzed and will be available at a later date. Below are some preliminary values whose range of accuracy is not established. These data are taken from reference 2.

<u>Specimen</u>	<u>E_T (watts/gm)</u>
Aluminum	0.63
Lead	0.48
Stainless Steel	0.85
Graphite	0.36

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Effects of Reactor Operation on Irradiation Facilities

There are six types of normal and abnormal reactor operations which may affect the environment about a test facility. The normal are reactor startup and shutdown, control rod movement, and changes in fuel exposure. The abnormal are xenon cycling and rare but possible massive water leaks.

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Startup: Each reactor has a standardized sequence for withdrawal of control rods. Thus, it is possible to predict the power distribution during startup and, if the specimen dimensions and flux requirements allow, place the specimen in a negligible flux gradient. During startup, a neutron flux gradient of 15 percent in two lattice units is possible; eight percent in two lattice units is normal. As seen in the Appendices, two lattice units are 15 inches in a K Reactor and 16.75 inches in C Reactor.

During startup, the total reactor power increases on a 60-second period until one percent of equilibrium is reached. From that point, the rate of power increase is two percent of equilibrium power per minute until 75 percent of equilibrium power is reached. At this point, localized regions of the reactor may be at equilibrium power and steps are taken to flatten the power distribution. The reactor power normally reaches 97 percent of equilibrium in six hours.

Reactor Shutdown: There are two types of shutdown, normal and scram. During a normal shutdown, reactor power is reduced to ten percent of equilibrium in 30 minutes by insertion of horizontal control rods, and is scrammed from that point. A scram shutdown occurs when a reactor is scrammed from equilibrium power. Figure IV shows the power transient during a scram.

Hanford reactors operate on irregular cycles, unscheduled outages occurring at unpredictable times. Past history has shown, however, that an 80 percent time operating efficiency is consistently achieved over a one-year period. A twenty-to-thirty day operating period between two-to-five day outages is normal.

Control Rod Movement: The side-to-side test facilities, which run parallel to the horizontal control rods, are at least three lattice units from any control rod. The maximum local effect that a specimen in a side-to-side facility would experience, due to any combination of horizontal control rod movement, is a 6.5 percent change in the thermal flux gradient in a distance of two lattice units.

The front-to-rear facilities are various distances from control rod positions. Appendix I shows front-to-rear flux traverses through positions one, two, and three lattice units from control rods.

The reactors have standardized equilibrium control rod patterns. Flux traverses can be obtained to identify very closely what flux shapes are available in a particular facility. Flux traverses may also be taken during a test to determine actual distributions. Also, the control rod pattern is flexible to some degree, and may be altered for testing purposes.

Fuel Exposure: The normal fuel cycle, for fuel in the central zone of the reactor, is two months. A process tube charged with green metal may operate as much as five percent lower than adjacent tubes charged with ripe metal. Normally, a fuel charge gains three percent in total tube power during one fuel cycle.

Xenon Cycling: Xenon cycles are movements of power generation from one portion of the reactor to another, with sixteen to twenty hours required for a complete cycle. It is a pessimistic view to expect two xenon cycles in a particular reactor per year, but it is possible. Some years are cycle free.

Approximately one-third of the significant xenon cycles experienced are severe enough to shut down part of a reactor. Normally, a change of 40 percent of equilibrium tube power is experienced between the maximum and minimum portions of the cycle.

During the worst xenon cycles, a power gradient of up to 20 percent in two lattice units is possible. Normally, a gradient of ten percent in two lattice units is experienced during a significant cycle.

The local effects of xenon cycles are not well known. Theoretical calculations show that the maximum possible flux gradient, due to xenon cycling, is 25 percent in two lattice units.

Water Leaks: The largest flux gradients ever seen in a Hanford reactor were caused by a massive water leak. A flux gradient of 25 percent in two lattice units was seen between wet and dry portions of the reactor. Only once in the history of the C and K Reactors has a water leak of this magnitude affected the testing facilities. Water leaks nearly always occur at the extreme front or rear of the reactor, far from any testing facilities.

In the K Reactors, one lattice unit below the bottom row of fuel, are four front-to-rear test facilities. Although the neutron flux available in these facilities is lower than that in the core, the flux distribution rarely deviates from a cosine.

Facility Coolant

The facilities are cooled by high quality water and the single pass cooling gives outlet temperatures averaging about 100 F. Coolant flows in the facilities normally range from 27 gpm to 34 gpm with standard orifice patterns; however, flows can be changed as necessary to accommodate capsules with a wide range of cooling requirements. Pressures in facilities can vary between 50 and 350 psig depending on the orificing and the pressure drop across the capsule. A laboratory is available for testing the hydraulic characteristics of experiments in facilities that duplicate the in-reactor installations.

Gaylord V. Guthrie
G. D. Guthrie, Physicist
Reactor Physics

References

1. P. J. Zimmerman, "In-Reactor Heat Generation and Encapsulation Criteria for Special Irradiations," DUN-689, September, 1966 (Secret).
2. P. J. Zimmerman, "Interim Report on In-Reactor Heat Generation Rate of Nonfissionable Materials," DUN-3619 (Secret).

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FIGURE I
FRONT-TO-REAR FACILITY TUBE CHARGED
WITH SOLID ALUMINUM

<u>REGION</u>	<u>MATERIAL</u>	<u>O.D. (IN.)</u>
1	Aluminum	2.500
2	Water	2.700
3	Zircaloy-2	3.260
4	Void	3.750 (Square)
5	Graphite	7.500 (Square)

FIGURE II
SELF-SHIELDING FACTOR (f_{ss}) FOR VARIOUS
GEOMETRIES OF EQUAL VOLUME

FIGURE IV
REACTOR SHUTDOWN BY SAFETY SYSTEM INSERTION

APPENDIX I

NEUTRON FLUX DISTRIBUTIONS

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FIGURE I
FRONT-TO-REAR NEUTRON FLUX DISTRIBUTION,
THREE LATTICE UNITS (22.5") FROM HORIZONTAL
CONTROL RODS (HCR'S)

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FIGURE II
FRONT-TO-REAR NEUTRON FLUX DISTRIBUTION,
TWO LATTICE UNITS (15.0") FROM
HORIZONTAL CONTROL RODS (HCR'S).

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FIGURE III
FRONT TO REAR NEUTRON FLUX DISTRIBUTION,
ONE LATTICE UNIT (7.5") FROM HORIZONTAL
CONTROL RODS (HCR'S).

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FIGURE IV
SIDE-TO-SIDE NEUTRON FLUX DISTRIBUTION

FIGURE V.

DEVIATION IN THE SIDE-TO-SIDE NEUTRON FLUX
DISTRIBUTION CAUSED BY LUMPED FUEL

Because of discrete lumps of fuel dispersed in the graphite in the side-to-side direction, a ripple is super-imposed on the side-to-side flux distribution. The ripple is as shown in Figure V, with the thermal flux suppressed near the fuel, and the fast flux at a maximum near the fuel.

The maximum amplitude of the thermal neutron flux ripple, (at the cell boundary) is plus or minus 5 percent for nearly any reactor loading. The fast flux ripple is normally plus or minus 6 percent, but can be larger for special reactor loadings.

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FIGURE VI

SIDE-TO-SIDE NEUTRON FLUX DISTRIBUTION
THROUGH REFLECTOR AND SHIELDING

APPENDIX II

LATTICE DESCRIPTIONS

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APPENDIX IIDescription of K-Reactor Natural and Enriched Cells

Natural cell (natural uranium in fuel region):

<u>Region</u>	<u>Material</u>	<u>OD (in.)</u>	<u>Density (gm/cc)</u>	<u>Physical Temp (°C)</u>
1	Water	0.4218	0.9913	57.4
2	Aluminum	0.5346	3.6207	103.0
3	Fuel	1.4373	17.5863	252.1
4	Aluminum	1.5343	2.7000	87.8
5	Water	1.7246	0.9917	56.4
6	Zircaloy-2	1.8075	6.4000	62.5
7	Void	1.9704	0.0000	500.0
8	Graphite	8.4628	1.6100	500.0

Enriched cell (0.947 wt. % U^{235} in fuel region):

<u>Region</u>	<u>Material</u>	<u>OD (in.)</u>	<u>Density (gm/cc)</u>	<u>Physical Temp (°C)</u>
1	Water	0.4338	0.9900	60.0
2	Aluminum	0.5468	2.7000	109.6
3	Fuel	1.4257	17.1676	256.5
4	Aluminum	1.5157	4.1521	90.9
5	Water	1.7245	0.9922	55.3
6	Zircaloy-2	1.8075	6.4000	61.8
7	Void	1.9704	0.0000	550.0
8	Graphite	8.4628	1.6100	550.0

The densities in regions 3 and 4 are not true densities. The overall lengths of natural and enriched fuel elements are 8.9 inches and 6.5 inches, respectively. The fuel lengths are 8.3 inches and 6.0 inches, and the end-cap effects are significant. To compensate for the end-caps, the end-cap volumes are added to the outer cladding by increasing the volume in region 4. The fuel density in region 3 is then decreased so that fuel fills the volume left by the end-caps without an increase in fuel mass.

Using the neutron fluxes listed for cells in an infinite lattice (Tables I and II), the following cross sections, for use in diffusion theory calculations, produce the correct reaction rates to generate 100 kw/ft in the fuel. They were calculated using the HAMMER, an eighty-four energy group transport code, and are homogenized over the cell.

Homogenized cross sections for a natural cell:

<u>Group</u>	<u>D</u>	<u>$\Sigma_a (x 10^{-13})$</u>	<u>$\nu \Sigma_f (x 10^{-3})$</u>	<u>$\Sigma_s (x 10^{-1})$</u>
1	2.3359	1.6189	3.6348	1.5547
2	1.0787	0.3842	0.0047	3.3226
3	1.0650	1.5152	0.5393	3.8846
4	0.8394	3.3156	3.8487	4.0074

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Total Macroscopic Scattering Matrix

	1	2	3	4
1	1.2691×10^{-1}	2.8545×10^{-2}	1.5716×10^{-5}	3.8295×10^{-10}
2	0.0000	3.2019×10^{-1}	1.2074×10^{-2}	1.7073×10^{-7}
3	0.0000	0.0000	3.7460×10^{-1}	1.3858×10^{-2}
4	0.0000	0.0000	2.7565×10^{-3}	3.9800×10^{-1}

Homogenized cross sections for a .947 percent enriched cell:

Group	D	$\Sigma_a (x 10^{-3})$	$v\Sigma_f (x 10^{-3})$	$\Sigma_s (x 10^{-1})$
1	2.3420	1.5610	3.5046	1.6034
2	1.0803	0.3756	0.0594	3.4518
3	1.0451	1.5820	0.6667	3.8906
4	0.8377	3.5520	4.5848	4.0160

Total Macroscopic Scattering Matrix

	1	2	3	4
1	1.4076×10^{-1}	2.8605×10^{-2}	1.6724×10^{-5}	4.0683×10^{-10}
2	0.0000	3.3517×10^{-1}	1.2186×10^{-2}	1.8103×10^{-7}
3	0.0000	0.0000	3.7534×10^{-1}	1.3716×10^{-2}
4	0.0000	0.0000	2.7949×10^{-3}	3.9879×10^{-1}

The streaming factor, due to all of the normally empty voids in a K Reactor, is 1.06. That is, the Migration Area (M^2) is:

$$M^2 = 1.06 (\tau + L^2)$$

This is essentially a leakage factor and does not affect the center of the reactor. In other words, while the streaming through the voids decreases the full pile k_{eff} , it has little effect on k_{∞} .

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FIGURE I
SINGLE NATURAL OR ENRICHED CELL

FIGURE II

REACTOR MODERATOR GRAPHITE STACK,
FRONT-TO-REAR VIEW.

FIGURE III
REACTOR MODERATOR GRAPHITE STACK,
SIDE-TO-SIDE VIEW

3A,B,C. - SPECIALTY IRRADIATION FACILITIES
HCR - HORIZONTAL CONTROL ROD POSITIONS

APPENDIX III

ENVIRONMENTS OF C-REACTOR FACILITIES

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APPENDIX III

The K Reactors and C Reactor are of the same conceptual design. All of the preceeding figures describing lattice cells are applicable to C Reactor except for their dimensions.

The following are descriptions of single natural and enriched fueled C-Reactor lattice cells:

Natural Cell (Natural uranium in fuel region)

<u>Region</u>	<u>Material</u>	<u>OD (in.)</u>	<u>Density (gm/cc)</u>	<u>Physical Temp. (°C)</u>
1	Water	0.3765	0.9890	61.9
2	Aluminum	0.4825	2.7000	103.8
3	Fuel	1.3771	17.5575	257.2
4	Aluminum	1.4731	3.6990	94.8
5	Water	1.6250	0.9860	67.4
6	Aluminum	1.7662	2.7000	69.7
7	Void	1.7940	0.0000	480.0
8	Graphite	9.4500	1.6320	480.0

Enriched Cell (0.947 wt% U-235 in fuel region)

<u>Region</u>	<u>Material</u>	<u>OD (in.)</u>	<u>Density (gm/cc)</u>	<u>Physical Temp. (°C)</u>
1	Water	0.3767	0.9881	63.6
2	Aluminum	0.4897	2.7000	108.0
3	Fuel	1.3769	17.3066	267.1
4	Aluminum	1.4669	3.9960	95.9
5	Water	1.6250	0.9865	66.5
6	Aluminum	1.7661	2.7000	69.0
7	Void	1.7950	0.0	500.0
8	Graphite	9.4500	1.0320	500.0

The peak power, in kilowatts per foot of fuel, is approximately the same for K and C Reactors and, therefore, the thermal flux is approximately the same. However, due to the larger graphite to fuel volume ratio in C Reactor, the fissions per cc are lower in C Reactor than in a K Reactor. As a result, the fast flux is approximately 30 percent lower in C Reactor than in a K Reactor.

The method given for calculating total heating of nonfissionable specimens is applicable to C Reactor.

The streaming factor, due to all of the normally empty voids in C Reactor, is 1.04. That is,

$$M^2 = 1.04 (\tau + L^2).$$

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The C-Reactor front-to-rear flux distribution is the same basic shape as that of a K Reactor. The fuel charge length is 23.6 feet with a 1.2 foot extrapolation length. This produces a 26.0 foot cosine distribution.

The C-Reactor side-to-side distribution is also the same basic shape as that of a K Reactor. However, due to the larger lattice cell, the ripple, superimposed on the side-to-side distribution, is larger than in a K Reactor. Actual measurements of this ripple have not been taken.

The effects of reactor operation on irradiation facilities are essentially the same at C and K Reactors. Two items of difference are: during startup, C Reactor reaches 97 percent of equilibrium power in one day; during a normal shutdown, C Reactor reaches ten percent of equilibrium power in 15 minutes.

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