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NEUTRON FLUX AND Cd RATIO MEASUREMENTS IN THE HN-1 BEAM HOLE FOR THREE FUEL LOADINGS OF THE OAK RIDGE RESEARCH REACTOR

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ABSTRACT

Neutron flux measurements were made in the Oak Ridge Research Reactor beam hole HN-1 shield plug, at low reactor power (N_L) with three fuel configurations. The purpose of these tests was to determine the most favorable fuel arrangement in the region of the experimental hole in order to permit minimization of exposure time of an in-pile slurry loop experiment using pure thoria.

It was found that the perturbed thermal neutron flux decreased by factors of 2 each 1.4 in. at the forward end of the beam hole. Maximum and average fluxes observed for three configurations were:

Fuel Configurations	Maximum Thermal Flux* (first inch)	Average Thermal Flux* (first 6 inches)
"High"	9.7×10^{13}	5.6×10^{13}
"Intermediate"	8.0×10^{13}	4.7×10^{13}
Present Operating	7.4×10^{13}	3.8×10^{13}

*Normalized to regular ORR power of 30 MW.

In the "high" and "intermediate" configurations fuel elements were located in the outer row of the lattice adjacent to the beam hole. Cadmium ratios were generally high (22-111) implying low available epi-cadmium flux under any of these configurations.

It has appeared possible that rearrangement of fuel elements in the Oak Ridge Research Reactor in the vicinity of horizontal beam hole HN-1 might result in a useful increase in thermal and epi-cadmium neutron flux in forward positions of this beam hole. An increased flux could materially decrease the exposure time for in-pile slurry loop experiment 0-1-28S.*

Flux-mapping experiments with three fuel configurations were consequently made to compare the flux that could be developed by alternate fuel arrangements.

The arrangement of the reactor core and experiment locations is shown in Figure 1. The layout of equipment in the vicinity of the beam hole HN-1 is shown in Figure 2.

In the three fuel arrangements only elements in positions C-8, C-9, D-8, D-9 and E-8, E-9 were changed. All other fuel elements in the array were fixed for the purposes of the test. In the "high flux" configuration, fuel was placed in positions C-9, D-9, E-9 and D-8. In the "intermediate" arrangement D-8 fuel was replaced with a Be moderator element, leaving fuel in positions C-9, D-9 and E-9. In the "operating" configuration, which has been the regular one during routine reactor operation, elements C-9 and E-9 adjacent to the reactor wall have contained Be, element D-9 directly in front of the HN-1 beam hole has contained an isotope stringer, and elements C-8, D-8 and E-8 contained fuel.

Figure 3 shows the fuel arrangement for the "high flux" set, which included tests 0-1-N_T-26 ("thermal" flux, unshielded Co) and 0-1-N_F-27 (Co enclosed in Cd). The fuel configuration for the "intermediate flux" set, including tests 0-1-N_T-28 and 0-1-N_F-29, is shown in Figure 4. The present "operating" configuration, shown in Figure 5, was associated with tests 0-1-N_T-30 and 0-1-N_F-31. Squares with two numbers refer to a fuel element

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*Memorandum from H. C. Savage, June 1, 1961, "Fuel Loading Change in ORR to Increase Neutron Flux in Beam Hole HN-1."

in the indicated lattice position, the upper number identifies the element, and the lower number gives the U-235 content in grams. Safety and control rod positions throughout the series are shown in Figure 6.

In Figure 7 a sketch of the nose of the HN-1 beam hole shield plug is shown. The flux monitors were inserted in the forward end of the center tube in the plug. The relative positions of the adjacent fuel element, designated D-9, the lattice face and the beam hole liner are indicated in the sketches. Also shown are the locations and thicknesses of the several water layers in the path of the flux.

Neutron flux measurements were made in the HN-1 beam hole of the ORR for the three fuel configurations on July 11 and 12, 1961. A 12-in. length of pure Co wire was exposed in the forward-most position in the HN-1 beam hole dummy plug for 10 minutes with the reactor at N_L (305 kw in this case). A second exposure under the same conditions was made with a similar wire encased in a Cd tube. This pair of exposures was repeated for the two other fuel configurations.

Table I presents the thermal neutron flux measurements based on a reactor power of 30.5 Mw. The Cd ratios determined in the three tests are listed in Table II.* Segment numbers shown in the tables refer to the position of the 1-in. length of wire that was counted; i.e., segment 1 refers to the piece extending from approximately 0 to 1 in. away from the end of the tube.

In Figure 8 the logarithm of the thermal neutron flux is plotted against segment position for the three fuel configurations. Over the first eight one-inch segments (corresponding to the water-shielded part of the core tube as shown in Figure 7) a fairly linear behavior is noted, the flux decreasing by a factor of two in approximately 1.4 inches in each case. The maximum thermal flux and the average flux over the first six inches (corresponding to the

*The computations, counting and consultation for all these measurements were performed by the Nuclear Analyses Group of the Analytical Chemistry Division (G. W. Leddicotte, group leader).

core region of in-pile slurry loop O-1-28S) for the three fuel arrangements are as follows:

Fuel Configuration:	"High"	"Intermediate"	Present Operating
Maximum Thermal Flux, first inch	9.7×10^{13}	8.0×10^{13}	7.4×10^{13}
Average Thermal Flux, first 6 inches	5.5×10^{13}	4.7×10^{13}	3.8×10^{13}

The cadmium ratios (activation of unshielded Co/activation of Cd shielded Co) were generally high (22-111), implying that in none of the configurations should an appreciable contribution from epi-cadmium neutrons be anticipated. The "high flux" configuration showed the lowest cadmium ratios, however.

The operating time of in-pile slurry loop experiment O-1-28S can be appreciably reduced by use of the "high flux" fuel configuration rather than the present arrangement. The loop will be loaded with pure thoria and is to be operated to achieve at least the fission fragment exposure of 7×10^{16} fissions/gram solids attained in the previous in-pile slurry loop experiment L-2-27S. It can be shown that under the proposed conditions total fissions are related to flux and time as follows:

$$\text{total fissions} \doteq K \phi^2 t^2$$

It is possible that the Oak Ridge Research Reactor will be operating at a power of 40 Mw rather than the present level of 30 Mw. On the basis of 40 Mw operation, operating times to achieve a minimum fission dose of 7×10^{16} fissions/gram solids have been estimated for the fuel configuration above. Realistic factors for the perturbation of the measured flux by the experiment have been used. At 40 Mw, with present operating fuel configuration a minimum operating time of over 14 weeks is required. With the "high flux" fuel configuration a saving of something over a month in operating time to produce the desired fission dose is estimated. Of course, ORR operation at less than 40 Mw power implies longer operation times in inverse proportion to power level.

It may well be desired to achieve higher doses in the experiment. In this case, substantially higher doses may be achieved with the "high flux" configuration in a given time, due to the dependence on the square of the flux.

The high values of the cadmium ratios in all configurations together with machine computations made previously suggest that the use of a beryllium reflector, as compared with graphite, immediately behind the core, offers little advantage in increasing the total dose to the loop core.

Table I. Thermal Flux

Segment No. (one inch)	Apparent Thermal Neutron Flux,* Normalized to ORR Power of 30.5 Mw (neutrons/cm^2- second) x 10^{-13}		
	"High Flux" Configuration Test 0-1-N_T-26	"Intermediate Flux" Configuration Test 0-1-N_T-28	"Present Operating" Configuration Test 0-1-N_T-30
1	9.72	8.03	7.39
2	7.66	6.43	5.12
3	5.84	5.00	3.77
4	4.40	3.76	2.81
5	3.28	2.81	2.00
6	2.40	2.06	1.40
7	1.62	1.39	0.90
8	1.05	0.90	0.46
9	0.83	0.66	0.40
10	0.72	0.56	0.36
11	0.63	0.53	0.33
12	0.66	0.84	0.30

*The average thermal flux of each segment is based on a cross section for Co^{59} of 36.7 barns and a half-life of 5.3 years for Co^{60}.

Table II. Cd Ratios*

Segment No. (one inch)	Cd Ratios* "High Flux" Configuration Test 0-1-N_T-26 and Test 0-1-N_F-27	"Intermediate Flux" Configuration Test 0-1-N_T-28 and Test 0-1-N_F-29	"Present Operating" Configuration Test 0-1-N_T-30 and Test 0-1-N_F-31
1	29.8	22.1	24.5
2	34.8	38.4	52.6
3	40.6	43.6	59.7
4	43.2	41.4	68.2
5	48.8	46.6	72.8
6	47.8	50.0	73.9
7	46.6	52.3	71.4
8	46.2	54.6	60.9
9	60.2	48.1	84.7
10	58.3	59.2	78.7
11	54.5	63.9	85.8
12	84.3	111.0	84.3

$$^{*}\text{Cd ratio} = \frac{\dfrac{Co^{59}\text{ bare wire, n/cm}^2\text{/sec}}{0.91\text{ (self shadowing correction)}}}{Co^{59}\text{ - Cd shielded wire, n/cm}^2\text{/sec}}$$

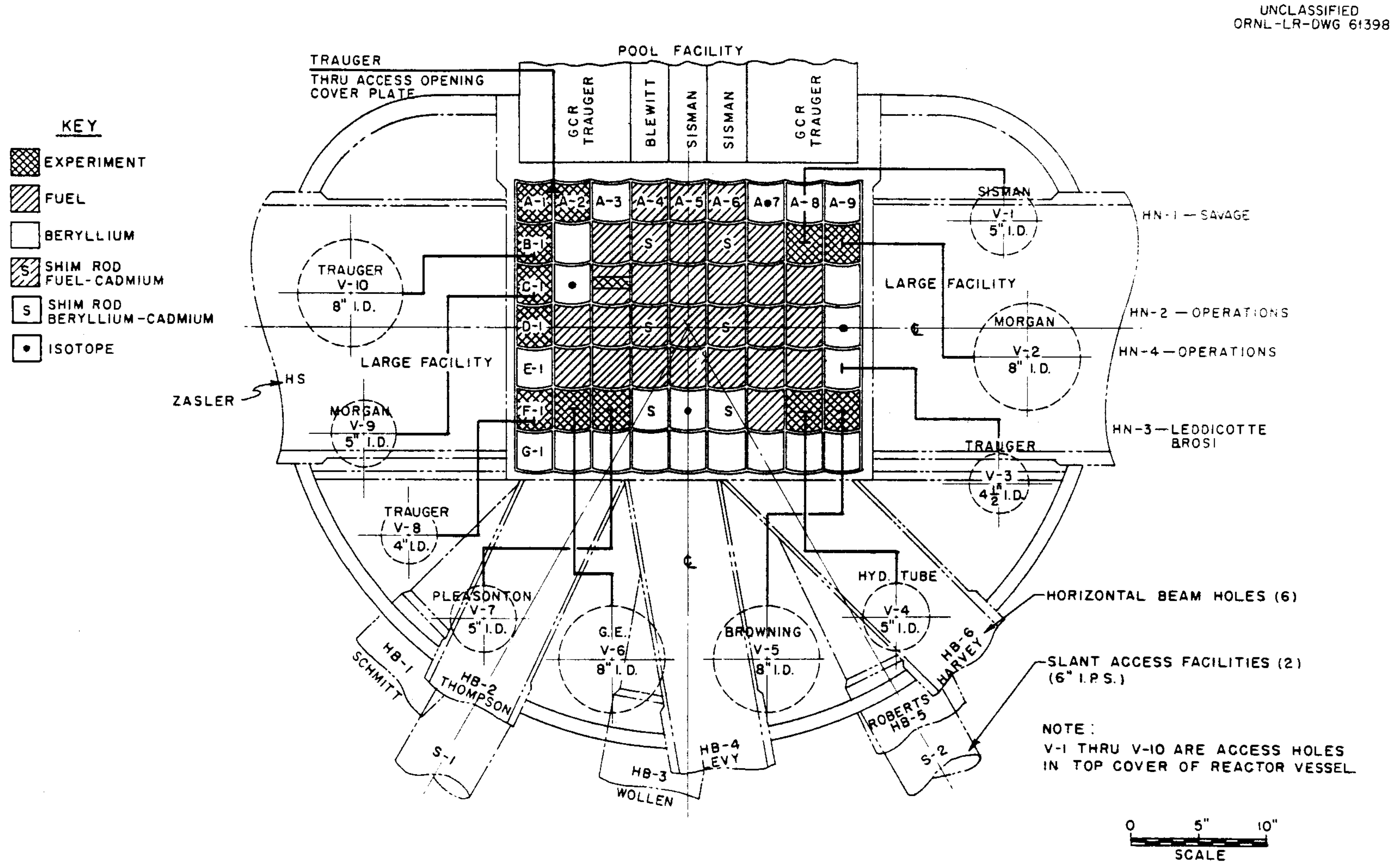

Fig. 1

O.R.N.L. RESEARCH REACTOR

LATTICE PATTERN AND EXPERIMENT LOCATIONS

RESEARCH ASSIGNMENTS & FUEL LOADING ON 6-30-61

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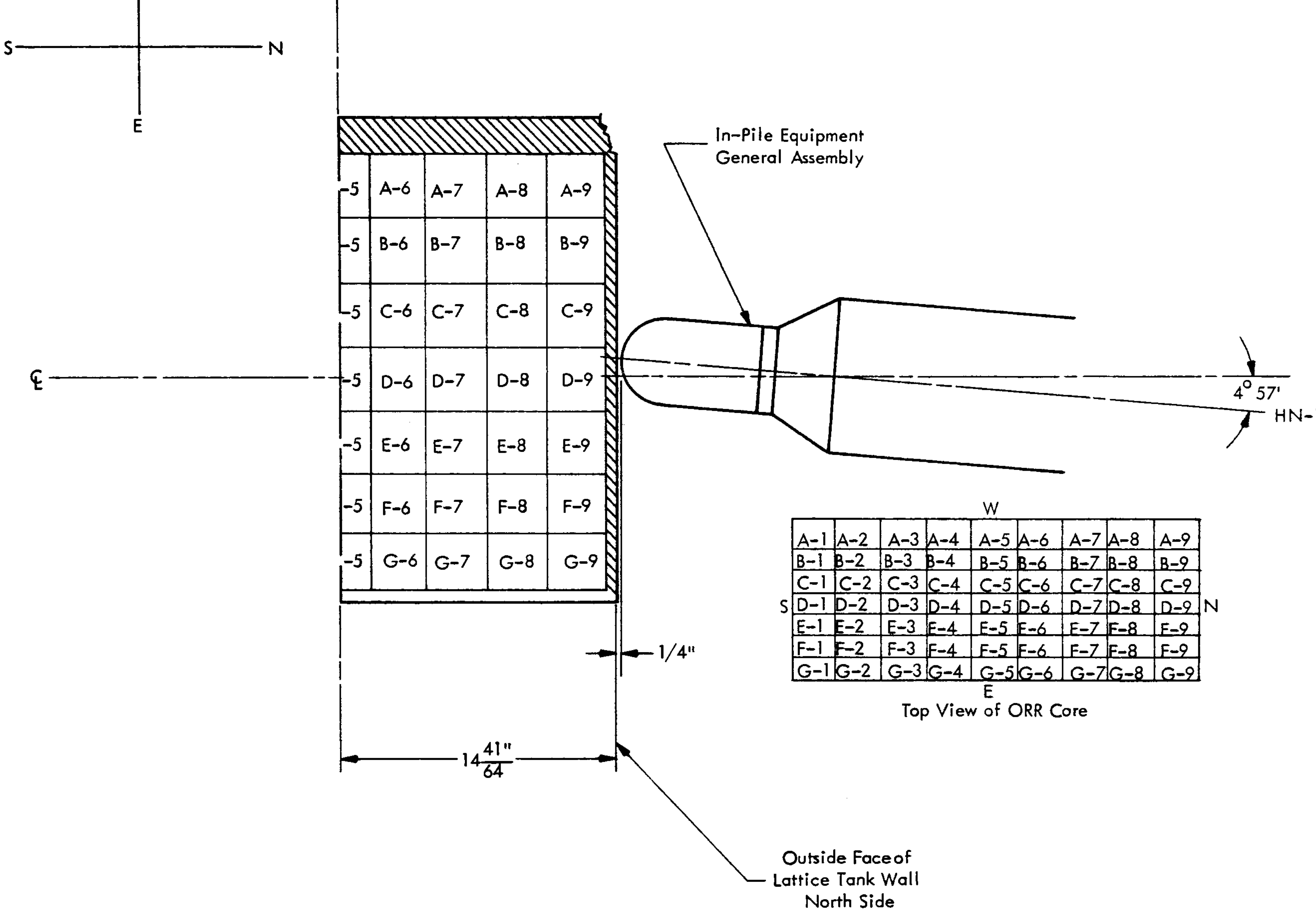

FIGURE 2. LAYOUT OF HN-1 EQUIPMENT

7-11-61

0-1-N_T-26

0-1-N_F-27

ORR CORE

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Unclassified

POOL

W

	1	2	3	4	5	6	7	8	9
A	A-1 MSR	A-2 MSR	A-3 I^{131}	A-4 4-A 165	A-5 366 177	A-6 303 164	A-7 I_{so}	A-8 Be	A-9 Be
B	B-1 GCR Loop 1	B-2 Be	B-3 325 192	B-4 255 79	B-5 285 150	B-6 265 93	B-7 302 165	B-8 Be Exp	B-9 Be
C	C-1 Fission Gas Exp	C-2 I_{so} Stringer	C-3 121	C-4 298 123	C-5 292 136	C-6 265 142	C-7 281 150	C-8 Be	C-9 341 177
S D	D-1 Be	D-2 380 195	D-3 350 172	D-4 273 731	D-5 269 138	D-6 285 131	D-7 7A 173	D-8 379 196	D-9 384 195 N
E	E-1 Be	E-2 324 177	E-3 374 195	E-4 337 169	E-5 373 196	E-6 328 168	E-7 378 196	E-8 Be	E-9 365 175
F	F-1 Be Exp	F-2 Exp	F-3 Ne^{23} Exp	F-4 Al Be	F-5 I_{so} String	F-6 Be 2-S	F-7 310 161	F-8 Hyd Tube	F-9 Meltdown Exp
G	G-1 Be	G-2 Be	G-3 Be	G-4 Be	G-5 Be	G-6 Be	G-7 Be	G-8 Be	G-9 Al

E

Cooling water outlet temperature: 88.5°C
Rod positions: 1 - at upper limit, 2 - at upper limit, 3, 4, 5 - at 15.255
Reference ion chamber: 0.72 x 2 x 10^{-7} amps
Count rate: 250 cps - 32" withdrawn

Figure 3. "High" Flux Configuration

7-11-61

0-1-N_T-28

0-1-N_F-29

ORR CORE

ORNL-LR-Dwg. 63952
Unclassified

POOL

W

	A-1 MSR	A-2 MSR	A-3 I^{131}	A-4 4-A 165	A-5 366 177	A-6 303 164	A-7 I_{so}	A-8 Be	A-9 Be	
	B-1 GCR Loop 1	B-2 Be	B-3 325 192	B-4 255 79	B-5 285 150	B-6 265 93	B-7 302 165	B-8 Be Exp.	B-9 Be	
	C-1 Fission Gas Exp	C-2 I_{so} Stringer	C-3 121	C-4 298 123	C-5 292 136	C-6 265 142	C-7 281 150	C-8 Be	C-9 341 177	
S	D-1 Be	D-2 380 195	D-3 350 172	D-4 273 731	D-5 269 138	D-6 285 131	D-7 7A 173	D-8 Be	D-9 384 195	N
	E-1 Be	E-2 324 177	E-3 374 195	E-4 337 169	E-5 373 196	E-6 328 168	E-7 378 196	E-8 Be	E-9 365 175	
	F-1 Be Exp	F-2 Exp	F-3 Ne^{23} Exp	F-4 Al Be	F-5 Iso String	F-6 Be 2-S	F-7 310 161	F-8 Hyd Tube	F-9 Meltdown Exp	
	G-1 Be	G-2 Be	G-3 Be	G-4 Be	G-5 Be	G-6 Be	G-7 Be	G-8 Be	G-9 Al	

E

Cooling water outlet temperature: 90°C
Rod positions: 1 - at upper limit, 2 - at upper limit, 3, 4, 5 - at 15.980
Reference ion chamber: 0.72 x 2 x 10-7 amps
Count rate: 160 cps - 34" withdrawn

Figure 4. "Intermediate" Flux Configuration

7-12-61
0-1-N_T-30
0-1-N_F-31

ORR CORE

ORNL-LR-Dwg. 63953
Unclassified

POOL
W

A-1 MSR	A-2 MSR	A-3 I^{131}	A-4 4-A 165	A-5 366 177	A-6 303 164	A-7 I_{so}	A-8 Be	A-9 Be
B-1 GCR Loop 1	B-2 Be	B-3 325 192	B-4 S 25S 79	B-5 285 150	B-6 S 26S 93	B-7 302 165	B-8 Be Exp	B-9 Be
C-1 Fission Gas Exp	C-2 I_{so} Stringer	C-3 121	C-4 298 123	C-5 292 136	C-6 265 142	C-7 281 150	C-8 384 195	C-9 Be
D-1 Be	D-2 380 195	D-3 350 172	D-4 S 27S 131	D-5 269 138	D-6 S 28S 131	D-7 7A 173	D-8 379 196	D-9 I_{so} Stringer
E-1 Be	E-2 324 177	E-3 374 195	E-4 337 169	E-5 373 196	E-6 328 168	E-7 378 196	E-8 341 177	E-9 Be Exp
F-1 Be Exp	F-2 Exp	F-3 Ne^{23} Exp	F-4 S Al(1-S) Be	F-5 I_{so} Stringer 2-S	F-6 S Be	F-7 310 161	F-8 Hyd Tube	F-9 Meltdown Exp
G-1 Be	G-2 Be	G-3 Be	G-4 Be	G-5 Be	G-6 Be	G-7 Be	G-8 Be	G-9 Al

S (left) — N (right)

E

Cooling water outlet temperature: 90°C
Rod positions: 1, 2 at upper limit; 3, 4, 5 at 15.100
Reference ion chamber: 0.725 x 2 x 10^{-7} amps
Count rate: 250 cps - 32" withdrawn

Figure 5. Operating Configuration

ORR CORE

ORNL-LR-Dwg. 63954
Unclassified

POOL
W

S

A-1	A-2	A-3	A-4	A-5	A-6	A-7	A-8	A-9
B-1	B-2	B-3	B-4 3_s	B-5	B-6 5_s	B-7	B-8	B-9
C-1	C-2	C-3	C-4	C-5	C-6	C-7	C-8	C-9
D-1	D-2	D-3	D-4 4_s	D-5	D-6 6c	D-7	D-8	D-9
E-1	E-2	E-3	E-4	E-5	E-6	E-7	E-8	E-9
F-1	F-2	F-3	F-4 2_s	F-5	F-6 1_s	F-7	F-8	F-9
G-1	G-2	G-3	G-4	G-5	G-6	G-7	G-8	G-9

N

E

6c actuated by servo control.

Figure 6. Safety and Control Rod Positions

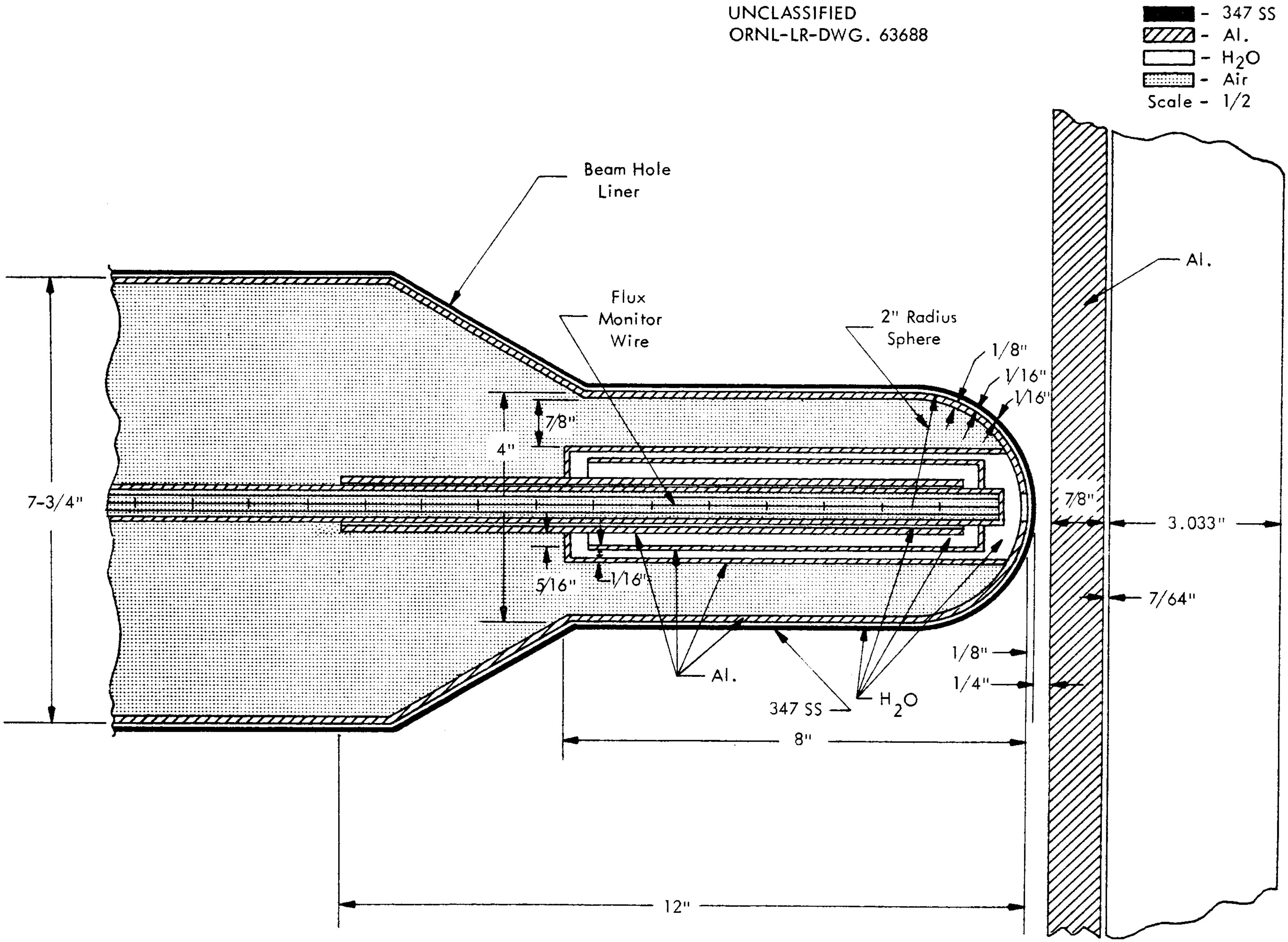

FIGURE 7. NOSE OF HN-1 DUMMY PLUG

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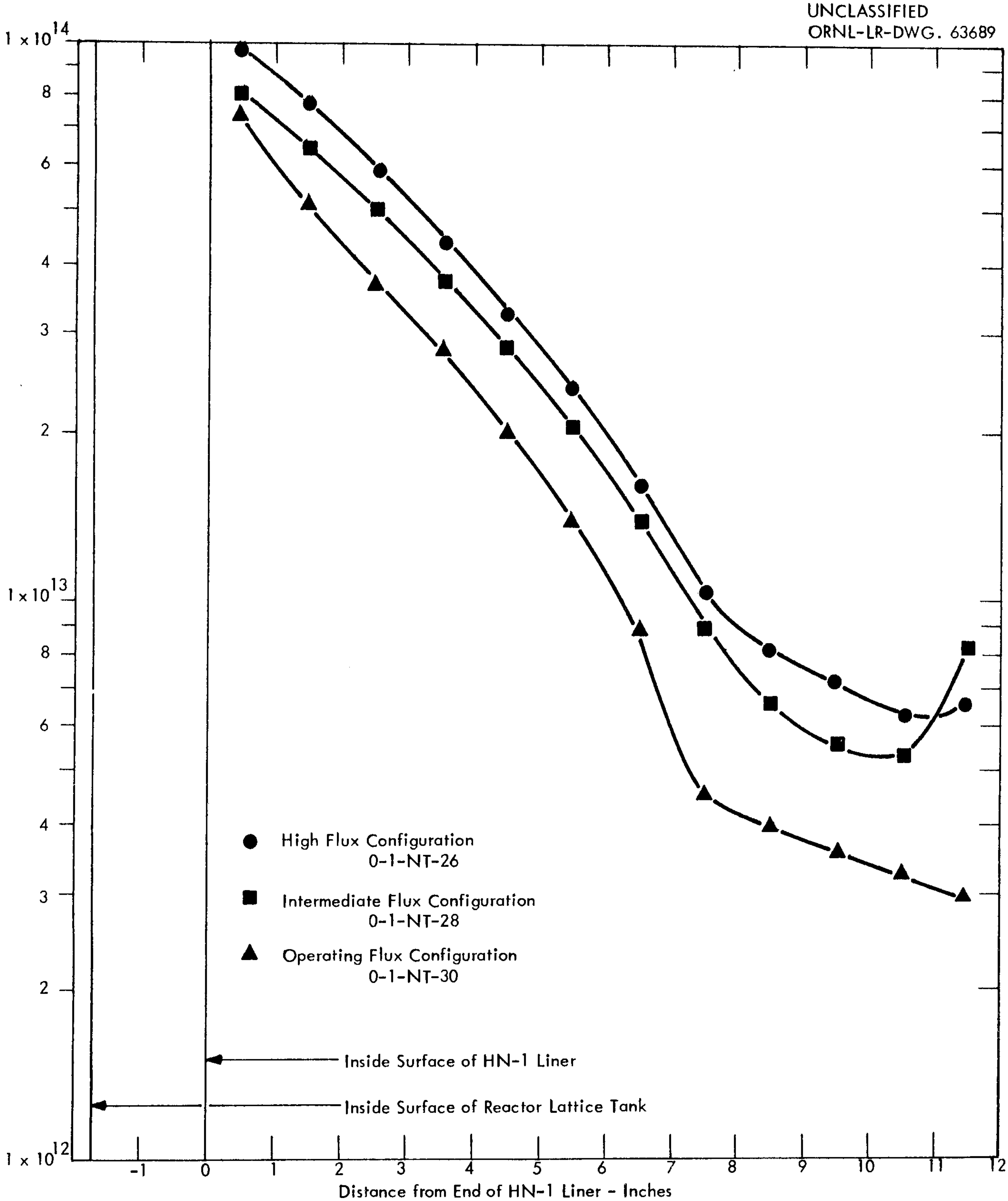

FIGURE 8. THERMAL NEUTRON FLUX MEASUREMENTS IN HN-1 FOR THREE ARRANGEMENTS OF ORR FUEL - REACTOR POWER = 30 MW

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