Westinghouse Advanced Reactors Division

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MASTER

RADIAL BLANKET AND SHIELD SIMULATION EXPERIMENTS
FIRST QUARTERLY PROGRESS REPORT FOR PERIOD ENDING

DECEMBER 31, 1971

RECEIVED BY TIC MAR 6 1972

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Prepared for the U. S. Atomic Energy Commission
Division of Reactor Development and Technology
Under Contract No. AT(11-1)-3045, Task 4A

Submitted to AEC/CHOO in February 1972

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SECTION 1

INTRODUCTION AND SUMMARY

1.0 GENERAL

This is the first quarterly progress report and covers the work performed during the second quarter of FY-1972.

Administrative control for the project was divided into the following sub-tasks:

RSAA-1000 Project Administration

RSAA-2000 Design and Fabrication of Shielding Simulation

RSAA-3000 Design of Spectrum Modifier and Pre-analysis of Experiment

RSAA-4000 Performance of Shielding Experiments

RSAA-5000 Establishment of Computational Capability and Analysis of the Experiments

1.1 SUMMARY OF OBJECTIVES

The objective of this program is to obtain sufficient technology to allow confident prediction of the shielding requirements for liquid metal fast breeder reactors (LMFBR).

To accomplish this objective, a simulation of the radial blanket, reflector, core barrel, and pressure vessel will be constructed. Neutron penetration through the simulation with an incident neutron spectrum representative of that in an LMFBR will be made. Analysis of the experimental results will be performed during FY-1973.

1.2 SUMMARY OF PRIOR WORK

This is a new investigation. No related work has been performed prior to the present period.

1.3 SUMMARY OF CURRENT PROGRESS

Work is proceeding on schedule per the 189a work plan.

Scheduling and purchasing studies have been made.

The material selection for the spectrum modifier has been completed subject to more detailed confirmatory calculations. Calculated neutron spectra for the proposed radial shield experiment have been compared with the corresponding spectra calculated for the Westinghouse LMFBR Demonstration Plant (Demo). The experimental simulation spectra are found to be in satisfactory agreement with the plant spectra.

Alternate materials to replace sodium in the radial blanket calandria of the simulation have been examined. This study was made to consider alternatives if future examinations find that sodium filling of the calandria is excessively difficult or expensive. Results of this study indicate that anhydrous Na_2CO_3 or atomized aluminum could be substituted for sodium in the blanket assemblies without seriously altering the neutron transmission through the blanket.

Layout drawings for the shielding test simulation and detailed layout drawings of the blanket assemblies have been completed and reviewed by the ORNL Neutron Physics Division staff. Preliminary manufacturing drawings of the radial shield components have also been completed and are currently being reviewed by Westinghouse Advanced Reactors Division (ARD) staff.

SECTION 2
RSAA-1000 PROJECT ADMINISTRATION
S. L. Davies, Jr., M. W. Dyos, E. Mullan

2.0 OBJECTIVES

The objectives of this task are to assure that the project is successfully completed, on schedule, within budget, and to the satisfaction of the Atomic Energy Commission; to assure compliance with contracted obligations; and to coordinate this project with other AEC-sponsored and Westinghouse-sponsored LMFBR development projects.

Overall project direction and day-to-day administration will be provided under this task. Plans and control will be established and maintained; periodic reviews will be held with the Commission; correspondences and reports will be coordinated; and day-to-day technical and administrative liaison with the Commission will be provided.

2.1 PRIOR WORK

Schedule 189a/Work Plan for FY-1972 was prepared and submitted to the Chicago Operations Office in November 1971. Technical work was initiated on the basis of this work plan.

2.2 CURRENT PROGRESS

For FY-1972 administrative control, the project is divided as outlined in Section 1.

Scheduling and purchasing studies were made to insure completion of this task within the allotted time period and funding limits.

Work is proceeding on schedule per the 189a work plan.

SECTION 3

TECHNICAL PROGRESS

3.0 GENERAL

In this section, the various subtasks of this project are reviewed, as follows:

- 3.1 RSAA-2000 Design and Fabrication of Shielding Simulation
- 3.2 RSAA-3000 Design of Spectrum Modifier and Pre-analysis of Experiment
- 3.3 RSAA-4000 Performance of Shielding Experiments
- 3.4 RSAA-5000 Establishment of Computational Capability and Analysis of the Experiments

3.1 RSAA-2000 DESIGN AND FABRICATION OF SHIELDING SIMULATION

M. W. Dyos, A. N. Krebs, N. C. Paik, J. F. Patterson, J. J. Ritts

3.1.1 OBJECTIVES

A simulation of the radial blanket will be designed and fabricated. The simulation will consist of depleted uranium oxide fuel pellets, clad in stainless steel tubes contained within sodium-filled tanks. Each tank will represent one row of radial blanket fuel assemblies. Stainless steel plates will be used to represent the reflector, core barrel, and pressure vessel. An inconel plate will be used to represent the core restraint springs, and tanks containing sodium will be used to represent sodium regions of the radial shield.

3.1.2 PRIOR WORK

This is a new investigation. No related work has been performed prior to the present period.

3.1.3 CURRENT PROGRESS

3.1.3.1 Considerations on Alternate Materials to Sodium for Blanket Assemblies

Initial considerations indicate potential difficulties and high cost in filling the radial blanket simulation assemblies with sodium. Pending further examination of this problem, alternate materials to replace sodium in the blanket have been examined. Primary considerations have been given to the use of sodium carbonate or aluminum as the substitute materials.

Neutronic calculations have been made with the one-dimensional neutron transport code, ANISN, using the source option. The one-dimensional model consisted of the spectrum modifier region with the neutron shell source at one side and eight regions which simulate radial blankets (~10 inches thick), radial reflector (~5 inches thick), radial springs (~5 inches thick), sodium gap (3 inches), core barrel (3 inches), sodium gap (3 inches), and finally reactor vessel (2 inches thick).

Approximately 50 percent theoretical density is thought to be the practical upper limit in packing a powder material of the sodium carbonate type into the blanket calandria. The calculated neutron flux spectrum is shown to be insensitive to the nominal density of packed powder within the void fraction available for a coolant material in the blanket calandria, as shown in Table 1.

Neutron transmission characteristics through the radial blanket assembly simulation are given in Table 2. The results show that transmission probability for total flux remains constant within one percent for all three cases with pure sodium, 25 percent packed sodium carbonate, and 40 percent packed aluminum.

Table 2 also shows that transmission probabilities for neutrons with a lower cut-off energy are in quite acceptable agreement (~5 percent) for these three cases. The transmission probability with a given lower cut-off energy is defined to be the ratio in percent of integrated neutron flux above the cut-off energy at the interface between the blanket and the reflector, to that at the interface between the spectrum modifier and the blanket. From results in Table 2 one finds that the transmission probability only increases about one percent as the density of either Na_2CO_3 or Al is increased by 10 percent. In fact, this is primarily because sodium or an alternate material with the volume fraction of 27 percent in the blanket contributes 10 percent or so to the macroscopic total cross sections in the region. Most of the macroscopic cross section is primarily determined by uranium, oxygen, and structural materials.

Organic and organometallic compounds of sodium have been ruled out from this study because of undesirable hydrogen atoms in the compounds. Various inorganic sodium compounds and aluminum powder have, therefore, been investigated as possible substitute material for pure sodium. Neutronic considerations examined for alternate materials were the effects of resonances, transmission probability, and distortions of the neutron spectrum in the blanket calandria and in the spectrum at the simulated core barrel location. Practical considerations of handling an alternate material, of procurement and of clean-up process following the experimental program, were also taken into account. From this study it is concluded that the most desirable substitution material based on neutronic consideration is sodium carbonate (25 percent packed powder of Na_2CO_3 , anhydrous). Major disadvantages in

Table 1. Normalized Neutron Spectra at the Radial Blanket and Reflector Interface

Group	E-Lower ^a	Pure Sodium Base Case (%)	Na ₂ CO ₃		Aluminum 40% Packed (%)
			25% Packed (%)	50% Packed (%)	
1	6.065 MeV	0.10	0.10	0.09	0.09
2	3.679	0.24	0.25	0.24	0.23
3	2.231	0.60	0.61	0.59	0.57
4	1.353	0.93	0.92	0.90	0.88
5	0.821 MeV	0.97	1.05	0.97	1.09
6	497.9 keV	4.68	4.56	4.16	4.67
7	387.7	1.49	1.46	1.34	1.52
8	302.0	2.31	2.27	2.01	2.35
9	183.2	6.79	6.75	6.15	6.86
10	111.1	7.79	7.71	7.13	7.51
11	67.38	9.63	9.56	8.96	9.32
12	40.87	7.64	7.63	7.29	7.59
13	25.54	8.49	8.35	8.11	8.20
14	19.89	5.57	5.47	5.34	5.74
15	15.03	4.34	4.29	4.28	4.52
16	9.12	6.94	6.97	7.13	7.24
17	5.53	5.18	5.37	5.58	5.51
18	3.35	4.90	5.86	5.77	6.66
19	2.84	0.55	0.70	0.67	0.91
20	2.404	0.54	0.68	0.65	0.84
21	2.035	1.22	1.36	1.40	1.46
22	1.234 keV	4.49	4.33	4.83	4.09
23	748.5 eV	3.32	3.09	3.60	2.77
24	454.0	2.62	2.43	2.89	2.14
25	275.4	2.12	1.97	2.37	1.73
26	130.1	2.64	2.46	2.99	2.16
27	61.4	1.60	1.51	1.83	1.33
28	13.7	1.54	1.49	1.78	1.33
29	0.68 eV	0.74	0.72	0.86	0.65

^aLower energy boundary for the group

Table 2. Percent Neutrons Transmitted Through Blanket

Material % Packing	Na 100%	Na ₂ CO ₃		Al	
		25%	50%	30%	40%
Neutrons with energy above					
1.00 MeV	6.1	6.3	5.4	6.3	6.0
100 keV	11.8	11.8	9.9	12.3	11.8
10 keV	17.4	17.3	14.9	18.0	17.3
1 keV	20.7	20.9	18.3	21.8	21.0
0.1 keV	22.7	22.7	20.3	23.3	22.6
Total	23.4	23.4	21.1	24.0	23.2

using sodium carbonate (or Na₂SO₄, Na₂SiO₄, and NaCl) are moisture pick-up by the compound, which results in undesirable neutron energy degradation by hydrogen atoms, and the extreme difficulty in handling a low density powder material. The use of sodium carbonate is recommended as the best alternate material, if and only if the tanks can be filled with the powder material without picking up moisture, and the tanks are completely sealed from moisture in the air during the experimental program. Although the detailed shape of the total cross section of aluminum does not bear a direct similarity to that of sodium, neutron slowing-down by aluminum does not significantly differ from that by sodium above the 2.85 keV resonance in the Na cross section. This study shows that aluminum (atomized particle with average size of 40 microns) can be used as an alternate material, in the event that the use of pure sodium or sodium carbonate is prohibited.

Finally, it is noted from this study that accomplishment of uniform density in filling out the blanket calandria tanks is not a significant problem as long as the specified density within ± 20 percent uncertainty can be realized.

3.1.3.2 Shielding Simulation Layout Drawings

Layout drawings were completed for the Demonstration Plant reactor core shielding test simulation. This simulation includes blanket assemblies, a number of solid plates, and the specially designed spectrum modifier stacked in front of the beam port of the TSR-II reactor (Tower Shielding Reactor at ORNL). The spectrum modifier, described in Paragraph 3.2.2.1, is designed to provide a neutron spectrum representative of that at the blanket interface of the Demonstration Plant.

The stacked assemblies and plates are designed to provide shielding effects which are representative of the core shielding components; namely, the radial restraint springs, sodium coolant, stainless steel core barrel, and stainless steel pressure vessel. A schematic of the proposed radial shielding

experiment arrangement is shown in Figure 1. Many of the plates to be used in this simulation are stock items available at the ORNL Tower Shielding Facility.

3.1.3.3 Radial Blanket Assembly Design

A detailed layout of a radial blanket shielding assembly, complete with 707 blanket fuel pins and suitable for filling with sodium, was completed. The assembly layouts were reviewed by the ORNL technical staff after which preliminary manufacturing drawings were completed, based upon their recommendations. The simulated radial blanket assembly is shown in Figure 2. A cost proposal for filling two blanket assemblies with sodium is being prepared by ORNL and will be submitted to the Westinghouse Advanced Reactors Division in early January 1972.

3.1.3.4 Manufacturing Drawings

Preliminary manufacturing drawings of all other shielding components and assemblies were also completed and are currently being reviewed internally at Westinghouse ARD. These drawings will be submitted to RDT and ORNL prior to proceeding with manufacturing.

3.2 RSAA-3000 DESIGN OF SPECTRUM MODIFIER AND PRE-ANALYSIS OF EXPERIMENT

M. W. Dyos, J. J. Ritts

3.2.1 OBJECTIVES

The materials required to convert the thermal reactor spectrum from Tower Shielding Facility into a spectrum similar to that at the core/blanket interface will be designed using nuclear analysis. A pre-analysis will be performed of the shielding configuration to analyze neutron penetration and spectral shifts through the simulation and to assess potential neutron streaming problems.

3.2.2 PRIOR WORK

This is a new investigation. No related work has been performed prior to the present period.

3.2.3 CURRENT PROGRESS

3.2.3.1 Spectrum Modifier Design

The spectrum modifier design has been completed subject to final, more detailed confirmatory calculations. The spectrum modifier is designed to alter the incident TSR-II beam port neutron flux spectrum (Figure 3) into one similar to that at the Demonstration Plant core/blanket interface (Figure 4).

Figure 1. Radial Shielding Simulation Schematic

Figure 2. Simulated Radial Blanket Assembly

Figure 3. TSR-II Source Spectrum

Figure 4. Demo Neutron Spectrum (Core/Blanket Interface)

The modifier design was limited to use of materials available at the ORNL Tower Shielding Facility. Many alternate materials for the modifier were analyzed to determine a material combination suitable for the desired spectrum modification. The final modifier configuration consists of the following materials:

1. Two inches of depleted ^{238}U (1-1/2-inch slab and 1/2-inch slab)
2. Two-inch slab of Type 304 stainless steel
3. Six-inch slab of sodium (available at ORNL canned in 1/4-inch walls of aluminum)
4. One-inch thickness of boral (two 1/2-inch slabs).

The calculational model used in the final analysis is shown in Figure 1.

In order to make results of individual calculations easier to compare, the 30-group flux results for each of the different modifiers were normalized to unity at the blanket-modifier interface. These results were then summed to give eight major groupings which were chosen as representative of important positions of the desired spectrum. The major peak in the spectrum was divided into four groupings with breaks chosen at resonances of materials likely to be present in the modifier (sodium, stainless steel, etc.).

Two important criteria were established in final selection of the modifier. The first criterion was that the general shape of the spectrum just inside the blanket simulation be similar to that just inside the Demonstration Plant blanket. Major peaks and valleys due to the resonance structure of the materials must appear in the spectrum.

The second criterion was to minimize the absolute differences (linear and squared) between the Demonstration Plant spectrum and the modified TSR-II spectrum based on the normalized eight group spectra discussed above. This minimization is given by the equations below.

$$L = \sum_{g=1,8} \left(F_d^g - F_m^g \right) \quad (1)$$

$$S = \sum_{g=1,8} \left(F_d^g - F_m^g \right)^2 F_d^g \quad (2)$$

where L and S are the linear and squared absolute difference to be minimized, respectively

F_d^g is the percent of the total flux in group g at a point just inside the Demonstration Plant blankets

F_m^g is the percent of the modified total flux in group g just inside the experimental test blankets.

The modified spectrum just inside the test blanket is shown in Figure 5. In the eight neutron energy groupings chosen (Table 3), the modified spectrum (final design) matches the desired spectrum of the Demonstration Plant with an absolute linear difference of 3.5 percent. The maximum difference of 1.6 percent occurs in Group 5. Table 3 also gives a comparison of the flux spectra for various modifiers that indicate the path followed in determining the final design.

3.2.3.2 Comparison of Neutron Spectra for the Simulation and the LMFBR Demonstration Plant

Calculations were made to determine the expected flux levels and spectra at locations which relate to specific positions in the Demonstration Plant blanket, shield, and vessel regions. A comparison of the spectra for these related positions is shown in Table 4 and Figures 6 and 7. The spectrum of the simulation matches the calculated Demonstration Plant spectrum fairly well behind the blanket, but diverges somewhat in Groups 5 and 6 between the reflectors and springs because of the absence of sodium in the reflector simulation. The flux shape recovers to a satisfactory matching in these groups at the front of the core barrel after the neutrons have passed through the 12-inch sodium slab.

3.3 RSAA-4000 PERFORMANCE OF SHIELDING EXPERIMENTS

M. W. Dyos

3.3.1 OBJECTIVE

The objective of this task is to measure neutron penetration of the radial blanket model.

3.3.2 PRIOR WORK

This is a new investigation. No related work has been performed prior to the present period.

3.3.3 CURRENT PROGRESS

A pre-analysis of the experiment has been made, and is described in Section 3.2.

This task is covered under separate funding, but liaison with the experiments will be conducted. No experimental work was scheduled in this period.

Figure 5. Experimental Simulation Spectrum (Modifier/Blanket Interface)

Table 3. Comparison of Various Spectrum Modifiers for Westinghouse Shielding Experiments

Grouping	Energy Limits	TSR-II Source Spectrum	Demo Plant Core-Blanket Interface Spectrum	Spectra for Various Modifiers ^(a) (Materials Placed in Order as Listed)									(Final Design) ^(b)
				4" SS	4" SS + 1" Boral	4" SS + 6" Sodium + 1" Boral	4" SS + 1" ²³⁵ U + 1" Boral	2.5" U + 1.5 SS + 6" Sodium + 0.5" Boral	1.5" U + 2" SS + 6" Sodium + 1" Boral	1.5" U + 2.5" SS + 6" Sodium + 1" Boral	2" U + 2" SS + 6" Sodium + 0.5" Boral	2" U + 2" SS + 6" Sodium + 1" Boral	
1	10 MeV-1.35 MeV	33.7	6.5	10.2	11.1	9.5	8.1	6.2	8.3	6.4	6.0	6.2	
2	1.35 MeV-0.39 MeV	10.6	15.5	23.4	24.4	17.5	22.7	14.5	15.9	14.8	14.7	14.8	
3	0.39 MeV-0.11 MeV	4.7	27.7	24.1	27.9	28.8	30.4	28.8	29.2	29.8	28.8	27.6	
4	0.11 MeV-25.5 keV	4.0	28.6	16.6	20.9	24.8	24.2	27.8	26.9	28.2	27.5	27.9	
5	25.5 keV-0.1 keV	2.7	12.3	8.6	9.2	11.4	9.4	12.6	12.1	12.8	12.6	13.9	
6	9.1 keV-203 keV	3.1	5.5	4.4	4.4	4.3	3.7	4.9	4.4	4.6	4.9	5.6	
7	2.03 keV-0.13 keV	4.6	3.9	10.3	2.0	3.5	1.4	5.0	3.1	3.2	5.2	3.9	
8	0.13 keV-0.25 keV	36.6	0.1	6.8	0.2	0.3	0.1	0.25	0.1	0.1	0.3	0.1	
Absolute Difference (See EQ. 1 - Linear)		-	Reference	45.1	26.6	12.6	23.1	5.4	7.5	5.5	5.9	3.5	
Absolute Difference Squared (See EQ. 2)		-	Reference	667	439	209	387	82.1	120	97.7	89.7	55.8	

(a) Spectrum values given as percentage of the total flux at the first ANISN meshpoint within the blanket.

(b) ANISN calculation on final design simulation; all others have regions similar to those in the Demonstration Plant beyond the blanket.

Table 4. Comparisons of Normalized Neutron Flux Values for the Demonstration Plant and the Westinghouse Shielding Experimental Simulation

Grouping	Neutron Energy Limits	TSR-II Beam Source Spectrum	Spectrum at Front of Blankets		Blanket-Reflector Interface		Reflector-Spring Interface		Front of Core Barrel	
			Demo Plant	Simulation With Modifier 2" U + 2" SS + 6" Na + 1" Boral	Demo Plant	Experimental Simulation	Demo Plant	Experimental Simulation	Demo Plant	Experimental Simulation
1	10 MeV-1.35 MeV	33.7 ^(a)	6.4	6.2	1.2	1.6	.6	.5	.3	.2
2	1.35 MeV-0.39 MeV	10.6	15.5	14.8	5.4	6.4	5.8	5.6	3.4	2.2
3	0.39 MeV-0.11 MeV	4.7	27.7	27.6	14.1	16.6	12.2	13.0	11.5	9.1
4	0.11 MeV-25.5 keV	4.0	28.6	27.9	23.3	25.8	18.7	16.5	18.5	14.9
5	25.5 keV-9.1 keV	2.7	12.3	13.9	15.0	17.3	6.3	11.3	10.8	10.6
6	9.1 keV-2.03 keV	3.1	5.5	5.6	15.8	13.8	10.8	16.7	5.5	5.5
7	2.03 keV-0.13 keV	4.6	3.9	3.9	19.4	14.9	24.2	23.3	24.4	29.1
8	0.13 keV-0.25 eV	36.6	0.1	0.1	5.8	3.6	21.4	13.2	25.7	28.4
Simulation Flux Reduction Factor (ϕ/ϕ_0 To Position Given)		1.0	-	0.32	-	0.06	-	0.025	-	0.002 ^(b)

(a) All flux values are given as percent of total flux at the point of interest.

(b) ϕ/ϕ_0 to the back of the simulation (5 inches of Type 304 stainless steel last zone for vessels) is approx. 2×10^{-4} .

Figure 6. Demo Versus Simulation Spectral Comparison at Blanket/Reflector Interface

Figure 7. Demo Versus Simulation Spectral Comparison at Front of the Core Barrel

3.4 RSAA-5000 ESTABLISHMENT OF COMPUTATIONAL CAPABILITY AND ANALYSIS OF THE EXPERIMENTS

M. W. Dyos

3.4.1 OBJECTIVES

The objective of this task is to develop the capability to perform detailed shielding analyses. To do this the two-dimensional discrete ordinates code, DOT, will be made operational, together with the supporting data generating codes.

The shielding experiments performed under Subtask RSAA-4000 will be analyzed during FY-73.

3.4.2 PRIOR WORK

This is a new investigation. No related work has been performed prior to the present period.

3.4.3 CURRENT PROGRESS

No work has been initiated in this period.