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ANALYSIS OF SAFETY-RELATED PHYSICS MEASUREMENTS IN THE INITIAL GCFR CRITICAL EXPERIMENTS

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ABSTRACT

Analyses of experiments in the initial critical assemblies for the gas-cooled fast breeder reactor (GCFR) were conducted using the calculational methods at General Atomic (GA) for GCFR design. The assemblies, constructed on the ZPR-9 facility at Argonne National Laboratory, simulated features of the 300-MW(e) GCFR demonstration reactor. Studies relating to the safety of this reactor design, and to the GCFR concept in general, were concerned with (1) reactivity coefficients of fuel and poison materials to evaluate loading and control requirements, (2) the worth of helium coolant in a depressurization event, (3) the Doppler effect in uranium to determine power coefficients, and (4) the effect of hypothesized steam ingress into coolant channels as a potential for reactivity addition and altering core neutronic and control characteristics. Results are reported for GA analyses of such safety-related physics measurements in two basic assemblies, the 3150-liter phase I core with a coolant void fraction of 55% and the 1300-liter phase II core with a 45% void fraction.

INTRODUCTION

The General Atomic (GA) program for developing the gas-cooled fast breeder reactor (GCFR) includes measurements of static physics parameters in zero-power critical facilities to validate GA design methods and nuclear data. The initial stage of the experiments was a three-phase, eighteen-month study on the ZPR-9 facility at Argonne National Laboratory (ANL), planned jointly by GA and ANL to provide data on fast reactor cores with significant void fractions. Several measurements in these critical assemblies relate directly or indirectly to the safe design of a GCFR. For these initial simulations, the goal of calculational/experimental comparisons is to gain confidence in predicting principal physics parameters such as

1. Critical loadings, fuel worth distributions, and zone enrichment.
2. Power distributions and peak-to-average power densities.
3. Worth of control materials, pointwise and in rod mock-ups.
4. Safety parameters such as Doppler and expansion coefficients.

5. Helium reactivity coefficient for depressurization analyses.
6. Neutron streaming and the effects of anisotropic diffusion.
7. Steam ingress effects, i.e., the worth of steam and its influence on control rod worths and temperature coefficients.

The designs of these critical assemblies and the analytical procedures used at GA are reviewed in the following sections. Results are presented for the analyses of experiments conducted in the first two phases which are relevant to the safety design features of the 300-MW(e) GCFR.

DESCRIPTION OF EXPERIMENTS

The reference design for these studies is a 300-MW(e) GCFR plant [1] having a 3200-liter multizoned core with an L/D of about 0.5 and an average enrichment of 17% fissile Pu in mixed oxide fuel rods. For the core and uranium oxide axial blankets, the volume fraction for the coolant (helium at 85 atm pressure) is 55%. In the radial blanket assemblies with fewer and larger uranium oxide rods, the coolant fraction is only 25%.

The fuel in the GCFR assemblies on ZPR-9 was simulated by combinations of Pu-U-Mo alloy, U_3O_8, and Fe_2O_3 plates inserted in rectangular drawers, and the "coolant" fraction was represented by voided rectangular sleeves of steel (calculations indicate that helium at 85 atm has little neutronic effect and is thus suitably represented by air). Details on the assemblies have been reported by Bhattacharyya et al. [2] and Bohn et al. [3].

The phase I GCFR assembly had a single-zone core of nearly the same volume, void fraction, and average enrichment as the reference design core, but with a higher L/D ratio to be accommodated in the ZPR-9 matrix. The purpose of the phase I studies was to obtain basic hard-spectrum physics information representative of helium-cooled, large GCFR cores. Additional experiments in phase I included scoping studies for simulated steam flooding and studies of neutron streaming in the coolant channels.

In the phase II core and axial blankets, 20% of the void channel per drawer was removed and the fuel materials spaced closer together, thus reducing the average void volume fraction to 45%, with the enrichment remaining at 17% in a higher concentration of fuel. The resulting phase II critical volume was 1300 liters, giving an L/D of about 1.05 for the 122.07-cm core length. The intent of these changes was twofold: (1) to investigate the effect of reduced leakage on calculational ability and (2) to provide a smaller system in which to conveniently conduct full-core steam flooding experiments. It was in this phase that several GCFR safety-related physics parameters were measured.

The third phase in this series of critical experiments utilized a three-zone core typical of the GCFR demonstration plant. Studies in this assembly include measurements of the effects of power flattening on important safety parameters and differences between plate and pin environments.

METHODS OF ANALYSIS

Basic to neutronics analysis for fast reactor studies at GA is the GGC-5 [4] computer program which determines the neutron spectrum on a fine energy mesh for specified regional compositions and prepares collapsed broad-group cross sections for use in diffusion or transport theory calculations. The input nuclear data tapes are prepared from ENDF/B-4 files. In the resolved resonance range, GGC-5 solves a two-region, slowing-down equation on a hyperfine grid to produce effective resonance integrals appropriate to the mixture, temperature, and heterogeneity specifications.

For the complex, multiplate ZPR-9 drawer cells, further corrections for heterogeneity effects are prepared using DTFX transport theory calculations in slab geometry. The DTFX-generated platewise and groupwise fluxes are used as flux advantage factors to produce the true cell reaction rates in each medium.

The structure of the GCFR fuel assemblies and the ZPR-9 simulations provide extensive axially oriented void channels through which neutron streaming occurs. To calculate the effect of preferential leakage in the neutronics diffusion theory programs, bidirectional modifications to the diffusion coefficients are required. An evaluation of such directional modifiers is provided by the Benoist theory [5], which is the basis of the computer programs PLADIF and PINDIF3 at GA for generating groupwise diffusion modifiers for plate and pin type cells, respectively. Sensitivity of neutronic behavior and, in particular, safety-related parameters, to the application of the bidirectional diffusion coefficients were investigated in the analyses.

Most of the neutronics calculations for these analyses utilized two-dimensional diffusion theory in ten energy groups, comparable to procedures employed in routine design calculations for the GCFR. Additional calculations with 28-group cross-sections were carried out for comparison and to judge the adequacy of the coarser structure for determining important safety parameters. The two-dimensional diffusion theory code 2DB [6] and the associated perturbation code PERT 5 were used, both of which were modified at GA to include the bidirectional diffusion modifier capability.

ANALYSIS OF BASIC NEUTRONICS

Eigenvalue results from the diffusion theory calculations on the basic configurations are shown in Table I. These values include the effects of cell heterogeneity and the diffusion modifiers which account for preferential axial leakage. Auxiliary studies carried out using standard isotropic diffusion coefficients resulted in eigenvalues which were 1.5% to 2.1% Δk higher, showing the strong leakage sensitivity. The good agreement in k (i.e., $k \approx 1.00$) lends confidence to the applicability of the Benoist approximation for reactors with significant fractions of void channels.

Calculations of fuel, control, and structural material worths at the center of the assemblies are compared in Table II with measured worths via calculated-to-experimental ratios (C/E values); the calculations were carried out with first-order perturbation theory. Ten- and 28-group results are reported for phase I; with the exception of polyethylene, the results for the two group structures are in close agreement. Similar results have been reported by Moore et al. [7] on the sensitivity of using directional diffusion coefficients for the real and adjoint flux calculations. These results clearly show the inadequacy of few-group first-order perturbation theory for computing hydrogen worth where extensive downscattering occurs. Table II also shows that the C/E discrepancies for most central worths in these GCFR assemblies are very similar to those reported for liquid metal cooled fast breeder reactor (LMFBR) assemblies.

TABLE I

Summary of Specifications for Initial GCFR Critical Experiments on ZPR-9

Assembly	Phase I	Phase II	Phase II Reflected	Phase III
Core volume, liters	3148	1300	1241	1911
Core radius, cm	90.60	58.23	56.88	70.60
Blanket radius, cm	--	82.76	82.76	99.59
Reflector radius, cm	--	--	96.22	111.38
Axial blanket thickness, cm	30.48	30.48	30.48	30.48
Axial reflector thickness, cm	--	--	13.46	13.46
Experimental reactivity, Ih [8]	+75 ± 2	+113 ± 1	+66 ± 1	
Calculated eigenvalues				
10-group RZ diffusion	1.00318	0.99789	0.99767	∿0.995
28-group RZ diffusion	1.00450	0.99803	0.99798	--

TABLE II

Analysis of Central Worth Measurements in the Initial GCFR Critical Assemblies using First-Order Perturbation Theory

Material	Core Center Reactivity Coefficients, Ih/kg				
	Phase I Assembly			Phase II Assembly	
	28-Group Calculation	10-Group Calculation	10-Group C/E	10-Group Calculation	10-Group C/E
CH_2	+23.7	-139	-1.41	-240.0	-1.56
He	-107.8	-108.4	--	-187.4	1.10
B-10	-2460	-2434	1.19	-3706	1.08
C	-20.55	-21.33	1.75	-35.50	1.67
Fe	-4.706	-4.879	1.45	-8.45	1.32
Th	--	-17.19	1.32	-26.6	1.24
U-233	--	--	--	346.7	1.17
U-235	--	127.1	1.28	203.4	1.21
U-238	-8.77	-8.69	1.32	-12.84	1.12
Pu-239	167.0	168.3	1.23	275.8	1.19
Pu-240	--	25.94	1.16	51.7	1.23

COOLANT DEPRESSURIZATION WORTH

A first-of-a-kind measurement in the phase II assembly was the determination of the central worth of helium, achieved by oscillation of empty and helium-pressurized metal cylinders into the center of the core. For a +15-cm to -15-cm axial extension, the measured helium worth was -162 ± 4 Ih/kg (with respect to void) [8]. First-order perturbation theory gave -180 Ih/kg and -183 Ih/kg using 10- and 28-group cross sections, respectively. Based on these results, the positive effect from depressurization or loss of helium in a GCFR reactor should be adequately predicted.

STEAM ENTRY STUDIES IN PHASE I

Steam entry into the coolant channels of a small central core zone in phase I was simulated by insertion of polyethylene foam slabs into the steel void sleeves. Various arrangements and perforations resulted in channel densities of 0.035, 0.0175, and 0.0088 g/cm^3 of CH_2, and measurements were made in a 5 x 5 drawer zone to depths of 30.5 cm in each assembly half. A detailed discussion of the analysis for this experiment is given by Hess et al. [9]; Fig. 1 shows the essential results of this study.

The measured CH_2 insertion worths in the phase I zone were positive and increased with channel density. The best analytical agreement in sign and value was obtained by complete regeneration of multigroup cross sections with the inclusion of spectrum moderation effects from hydrogen; this procedure has always been recognized as a necessity, and is routinely used, for steam ingress safety studies for the GCFR demonstration plant. The use of exact perturbation theory, or well converged k calculations, and appropriately reaveraged cross sections gave C/E values for the 10- and 28-group analyses of about 0.95 and 1.10, respectively, for the worth of CH_2 in the central zone. The higher C/E value is comparable to those for first-order perturbation results for non-hydrogenous materials.

STEAM ENTRY STUDIES IN PHASE II

Full-core and blanket steam ingress was simulated in the phase II reactor using polyethylene foam. A maximum channel density of about 0.0175 g/cm^3 was achieved with perforated foam slabs inserted into all void channels. These experiments were carried out in the reflected phase II configuration, but no CH_2 additions were made in the reflector regions. A sequence of loadings was adopted to yield steam worth data under a variety of conditions; of primary importance was the effect of poison rod loadings upon steam worth, a situation directly relevant to conditions in an operating reactor. Table III lists the assembly configurations for which CH_2 worths were measured. As in the Phase I scoping studies, the worths of CH_2 were positive for all densities and configurations examined. Agreement between calculations and experiment is not as good as for the small central-zone measurements in phase I; the calculational discrepancy can most likely be traced to the diffusion parameters for the core and blankets, since leakage played a lesser role in the central-zone study.

Encouraging results of the analyses include the qualitative agreement with experiments for the negative effect on the flooding reactivity worth caused by control rod installations and increased core dimensions. It should be stressed that in the power reactor some poison will be in the core under all conditions, either as control rods or fission products. In

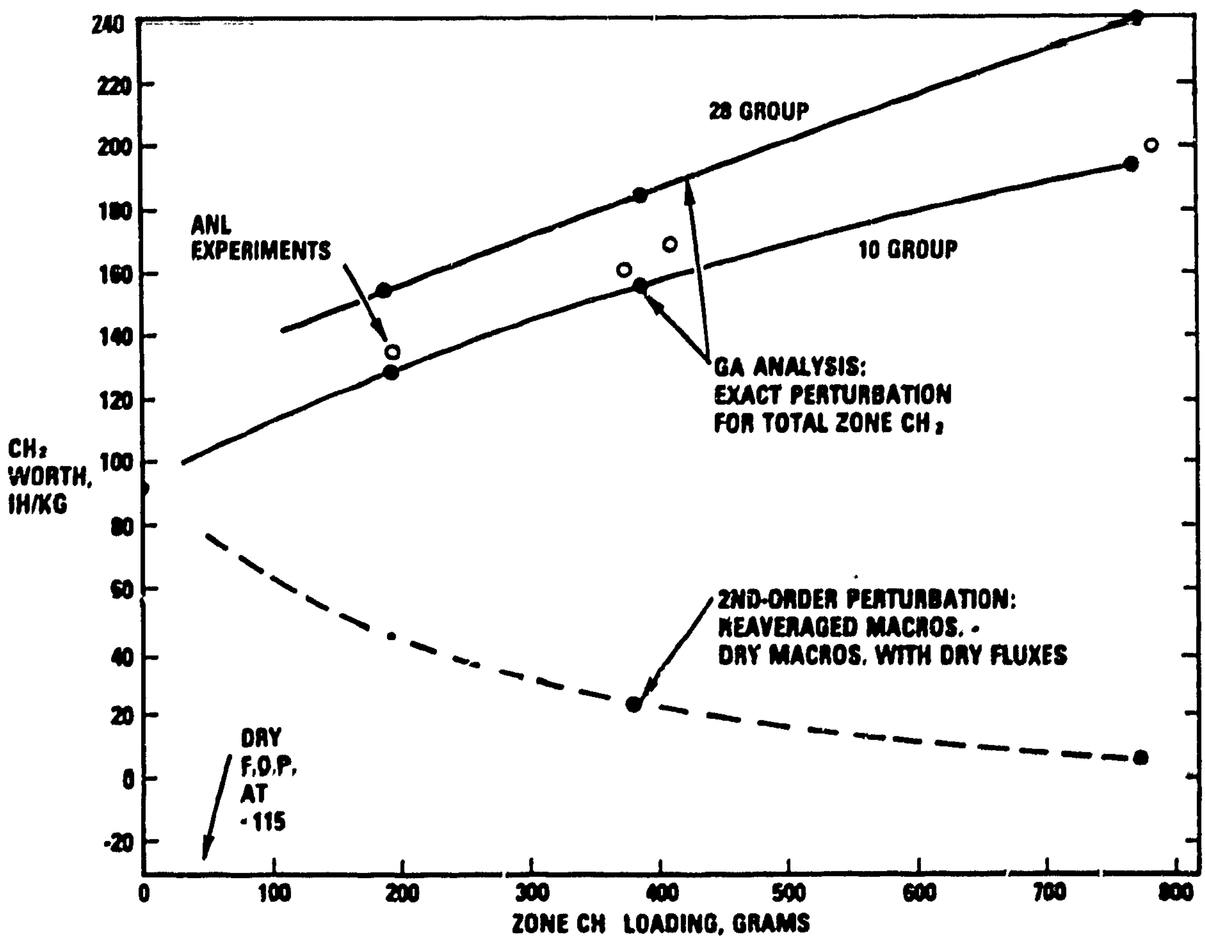

Fig. 1. Analysis of steam flooding worths in Phase I central zone

TABLE III

Analysis of Steam Flooding Experiments in Full Core and Blankets of Phase II GCFR Critical Assembly

Assembly Configuration		Flooding Worths Calculated by Δk Difference, RZ Diffusion Calculation			
Core Radius, cm	Simulated B_4C Control Rods Installed	10-Group Analysis		28-Group Analysis	
		Ih	C/E	Ih	C/E
54.79	None	1096.6	2.08	1218	2.31
56.10	None	1017.3	2.08	1135	2.32
56.10	1 at core center	922.4	2.15	1049	2.45
56.10	Center + 7-rod ring	396.4	1.10	545	1.51
59.38	Center + 7-rod ring	246.0	1.21	378.6	1.86

addition, the 28- versus 10-group comparison for phase II shows the same increased worths for the finer structure as was found in the phase I analysis.

In addition, this study establishes an upper limit for reactivity insertion due to steam ingress into an actual GCFR power reactor. The phase II configuration represents a very pessimistic situation regarding steam flooding: (1) it has a small core (1300 liters) and less than optimum blanket and reflector thicknesses, both of which accentuate the positive leakage component of steam worth; (2) the core contains no poisoning by control rods or fission products, a condition which would never exist in an actual power reactor; (3) the critical assembly fuel operates at room temperature (300 versus ~1300 K in the power core) with reduced capture rates in uranium. Thus, a positive 500 inhour value is a very conservative upper limit for the worth of steam ingress at 0.0225 g/cm^3 H_2O (hydrogen density as in 0.0175 g/cm^3 CH_2), a density which itself is a highly pessimistic assessment of the potential ingress from a steam generator leak.

CONTROL ROD WORTH STUDIES

The phase II program included simulations of B_4C control rod insertions in combination with the sequence of loadings involved in the steam worth measurements. The rods consisted of core-length columns of clad natural boron carbide installed in place of 1.27-cm-wide columns of void sleeves. Per individual rod the substitution provided a B-10 loading of about 220 g with minor net changes in steel content. The rod locations included a drawer at the core center and seven drawers distributed in a ring with an average radius of about 70% of the core radius; the objective was fairly uniform core poisoning.

Center rod worths and center-plus-ring rod worths were measured with and without simulated core steam flooding. Table IV lists the results of the ANL measurements along with values given by the GA 10- and 28-group analyses. Cross sections for these calculations were generated in the dry and flooded cell GGC-5 runs with the B-10 and B-11 materials at infinite dilution. Thus, the shielding or flux factors for the boron are not representative of a true rod mock-up. Detailed self-shielding calculations for this B_4C column are under way, and the high C/E values (1.10 + 1.13) should be reduced by a proper evaluation.

TABLE IV

Analysis of Simulated B_4C Control Rod Worths in Phase II GCFR Critical Assembly With and Without Steam Ingress

CH_2 Density in Void Channels, g/cm^3	B_4C Rods Installed (1/4 x 2 x 48-in. columns)	Worth of Installed B_4C Rods, Ih			
			2DB Diffusion Calculations		
		Experiment [8]	10-Group	28-Group	C/E, 28-Group
None	1 at core center	-484.2 ± 1.9	-533.9	-532.6	1.100
None	Core center + 7-rod ring	-1934 ± 35	-2168.3	-2176.8	1.126
0.0175	1 at core center	1544.8 ± 3.7	-628.9	-619.1	1.136
0.0175	Core center + 7-rod ring	-2061 ± 47	-2798.3	-2769.0	1.343

TABLE V

Analysis of U-238 Doppler Effect Measurements at Center of Phase II GCFR Critical Assembly

Configuration	Measured Doppler Effect in Depleted U for ΔT = 300 to 1100 K, Ih/kg [8]	300 to 1100 K Doppler Effect in U-238 Calculated by First-Order Perturbation Theory			
		28-Group Analysis		10-Group Analysis	
		Ih/kg[a]	C/E	Ih/kg[a]	C/E
Reference dry core, core radius = 56.88 cm	-0.623 ± 0.009	-0.654	1.050	-0.612	0.982
Flooded case, 0.0175 g/cm^3 CH_2 in core and blanket void channels, core radius = 56.10 cm	-1.200 ± 0.010	-1.302	1.085	-1.507	1.256

[a]Using conversion factor of 942 Ih/%k.

The high C/E for the analysis of the eight-rod worths in the flooded case may be a result of cross section averaging with an incorrect spectrum or modeling of a symmetric annular ring for the seven outer rods which actually were in an asymmetric pattern. Exact modeling in XY geometry will be needed to clarify this discrepancy.

DOPPLER COEFFICIENT ANALYSIS

The Doppler effect in uranium over a temperature range of 300 to 1100 K was measured at the center of the phase II core in a dry and a CH_2-flooded configuration. The UO_2 sample used was about 2.5 cm in diameter and 30.5 cm long, giving a measured effect over ±15 cm from the midplane. An exact modeling of this sample in spectrum and transport cell calculations has not yet been carried out. However, the GGC-5 cases for the U_3O_8 plate in the phase II core cell with and without CH_2 were rerun with the U_3O_8 at 1100 K. Table V compares the analysis for the 300 to 1100 K Doppler effect in U-238 using the hot U_3O_8 cross sections with the ANL measurements for this total temperature change. Both group structures give reasonable predictions for the Doppler effect in the dry case, but the effect in the steam-flooded configuration is better calculated using the finer group structure.

It can therefore be seen that the analytical methods with 28-group cross sections adequately project the increase in absorption in U-238 with increasing temperature and also with a softer spectrum caused by ingress of hydrogeneous material. This should lend confidence that the steam worth calculations for a hot reactor have the same validity as the analysis for the critical experiment floodings.

CONCLUSIONS

Among the general conclusions reached in the GA analyses of the GCFR critical experiments to date are the following:

1. Fuel and cladding worths can be predicted with approximately the same confidence for the GCFR as for the LMFBR; i.e., there are similar bias factors.

2. The worth for helium depressurization in a GCFR is adequately predicted.

3. Individual control rod worths appear to be predicted reasonably well by eigenvalue difference techniques, although more rigorous analyses as well as studies of rod banks and rod interactions in future GCFR critical assemblies are needed.

4. Accurate calculation of steam ingress effects requires that careful attention be paid to energy mesh, spectrum averaging, and leakage parameters. Investigations of the methods involved are needed to reduce the calculational discrepancy for full-core flooding.

5. Steam entry into a hot, poisoned GCFR, expected to be a negative worth, will increase the negative prompt temperature feedback coefficient via the increased U-238 Doppler effect, thus enhancing safety performance. Conversely, however, the hot-to-cold shutdown rod worth requirements are slightly greater.

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