SUMMARY REVIEW
OF THE
KINETICS EXPERIMENTS ON WATER BOILERS

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ABSTRACT

With the objective of providing information for aqueous homogeneous reactor safety and hazards evaluations, experimental data and analysis of the kinetic behavior of five core loadings studied in the Kinetic Experiments on Water Boilers (KEWB) is presented. The experimental data covers the reactor response to ramp reactivity insertions for ramp rates up to $0.16/\text{sec}$ with total reactivity inputs up to 6.25 , and step reactivity insertions up to 5.50 . The core loadings range from solution volumes of 11.5 to 26.0 ℓ and fuel concentrations from 57 to 203 gm of U^{235}/ℓ . A range of periods was investigated extending below 0.6 μsec and to peak powers in excess of 4000 Mw. The only safety limitation found in these experiments was an inertial pressure resulting from the production of radiolytic gas. The pressure was characterized as a function of reactor parameters by a series of experiments up to 620 psi.

An analytic model of aqueous homogeneous reactor kinetics is presented. This model consists of the space independent kinetics equations with six groups of delayed neutrons, temperature compensation of reactivity with a quadratic coefficient in temperature, reflector moderated neutrons treated as a seventh delay group, and radiolytic gas void compensation of reactivity according to the model of aqueous homogeneous reactor void growth. An analytical expression was developed to predict peak pressure as a function of reactor parameters. Bursts computed with these models and experimental data from four KEWB reactor core loadings are in excellent agreement.

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I. GENERAL INTRODUCTION

This document has been divided into two parts, roughly covering the experimental and analytic phases of the KEWB program. Although dealing largely with the specific results of the program, the material is applicable to the kinetics, safety characteristics, and hazards evaluations of aqueous homogeneous reactor assemblies in general. It is intended for use in conjunction with three other KEWB reports^{1,2,3} and for those desiring to perform computations, a computer program.⁴

II. SUMMARY OF THE EXPERIMENTAL PROGRAM – R. K. STITT

A. INTRODUCTION

The general objective of the Kinetic Experiments on Water Boilers (KEWB) program was the determination of the safety characteristics of small aqueous homogeneous reactors known by the title of "Water Boilers." These are typically small research and training reactors of less than 100 kw steady state, which are fueled with 10 to 30 l of an aqueous uranium solution. They normally operate at temperatures below the boiling point of water and absolute pressures of less than 1 atm.

The desire to locate these reactors in highly populated areas justified an investigation of their kinetic behavior under abnormal operating conditions. Since release of fission products to the environs is the matter of greatest concern, the failure of the core vessel due to internal pressure is the criterion upon which the safety of solution reactors must be judged.

This consideration led to an emphasis on the measurement and theoretical understanding of the origin and nature of pressure waves generated within the KEWB test reactors during large power excursions.

B. DESCRIPTION OF THE KEWB EXPERIMENTAL FACILITIES

The site selected for these tests is at the Atomics International Field Test Laboratory about 25 miles northwest of central Los Angeles at an elevation of 1800 ft in the Santa Susana mountains.

The test building, which housed the reactor and associated plumbing is an underground concrete building consisting of a 12- by 15-ft reactor room, a 6.5- by 7.5-ft auxiliary apparatus room, and a 6.5- by 7.5-ft operations gallery. The entire structure was covered with 6 ft of earth for shielding purposes. Since the reactor was to be operated on large power excursions, the control room from which all tests were initiated and recorded was constructed 200 ft away to provide additional radiation shielding for operating personnel during these power bursts. A detailed description of the facility is given in an earlier report.¹

C. REACTOR CORES AND LOADINGS TESTED

1. "A" Core

The first core vessel installed and tested in the facility, referred to as the "A" core, was a prototype of an existing 50-kw reactor in all dimensions except shell thickness. The increase in shell thickness was intended to permit the examination of conditions which might rupture a thinner vessel. Since destructive tests were not planned, extreme conditions could be investigated without danger of core vessel rupture.

A layout drawing and photograph of the KEWB "A" core vessel are shown in Figure 1. The sphere had a 12.3 in. ID and a minimum shell thickness of 0.22 in. The test pressure was 1500 psi with the yield point calculated at 2200 psi. The overflow chamber attached to the top of the sphere consisted of a central 2-in. tube 18 in. long enclosed in a 4.5-in. outer tube. The deflection

Figure 1a. Layout Drawing of "A" Core

Figure 1b. Photograph of "A" Core

dome at the top of the central tube aided in the separation of the gas and liquid during excursions which expelled fuel solution from the sphere into this overflow chamber. The fuel solution was deflected into the annular region between the tubes and retained there. The hydrogen-oxygen gases passed through a stainless steel entrainment eliminator to the recombiner or ballast tank. The fuel solution drained back to the core from the overflow chamber through a 3/16-in. hole.

A 1.6-in. ID tube passed through the sphere along a horizontal diameter. Four additional tubes, which passed through the sphere parallel to the vertical diameter, contained the four normal control rods. The spherical part of the vessel contained three bosses for pressure transducers and two bosses for thermocouples.

Cooling of the fuel solution was provided by cooling coils of stainless steel tubing 0.25-in. OD with 0.028-in. wall. Ten coils, each 9 ft long, were connected to external inlet and outlet manifolds along one side of the core. This provided cooling adequate to maintain the fuel solution at less than 80°C when the reactor was operating at 50 kw.

The control rod system consisted of four normal sintered B_4C rods contained in stainless steel thimbles. The poison section was 16 in. long and 0.75 in. in diameter. The rods were positioned by conventional rack and pinion drives and fell free when released by electromagnets which attached them to their individual racks. The rods controlled about \$2.1 of reactivity each and a total of \$9.0. The withdrawal rate for the rods was 6 in./min, which corresponds to \$0.04 $\Delta\rho$ /sec/rod for the linear portion of the control rod calibration curve.

The reflector consisted of unmachined AGOT-grade graphite bars stacked to form a 56-in. cube. The rounded edges of the unmachined bars left small voids along each edge when stacked. This reduced the gross density of the reflector to less than the normal 1.6 gm/cm³ by a few percent. The graphite bars that were adjacent to the core were machined to fit the shape of the vessel. The reflector and core were enclosed in a reinforced aluminum secondary enclosure.

2. "B" Core

The second-core vessel installed and tested in the KEWB facility, referred to as the "B" core, is of cylindrical geometry. A layout drawing and photograph of the "B" core is shown in Figure 2. The cylinder is 12 in. in diameter with an overall length of 36 in.

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Figure 2a. Layout Drawing of "B" Core Figure 2b. Photograph of "B" Core

The vessel was tested at 1250 psi and had a calculated yield strength of 1800 psi, based upon the measured yield strength of the material actually used in its construction. The fuel solution occupied the lower 12 to 16 in. of the vessel with a void of more than 1 ft above it. The change of cores to one of cylindrical design was necessary if the fundamental pressure wave, resulting from the rapid production of radiolytic gas, was to be fully observed. Reflected pressure impulses and gradients brought about by the neck constriction of the

spherical core prevented such observation. The cylindrical design allows a constant free surface area for fuel expansion during reactor excursions. This core was not equipped with an effluent tube nor holdup reservoir. The deflection dome near the top serves to prevent expulsion of the fuel into the associated plumbing system.

Transient experiments with this core showed that following a large burst it was unnecessary to trap the fuel solution, even temporarily, in a liquid overflow vessel to prevent reoccurrence of the transient. So much gas was produced during the excursion that by the time the two phases separated and the liquid had become quiescent, the solution was at boiling temperature due to the power plateau following the burst.

The vessel was fabricated from 0.25-in. stainless steel material with standard elliptical stainless steel pipe caps forming the ends. A 1.99-in. ID tube passes through the cylindrical part of the vessel on a horizontal diameter 6 in. from the bottom of the vessel. This tube accommodates the poison rod which is used to initiate power transients. The diameter of this tube was made bigger than the corresponding tube in the "A" core to allow for a larger poison rod needed to initiate the faster transients planned with this core. Four 1-in. tubes pass through the entire length of the vessel, parallel to the vertical axis of the cylinder. These four tubes contain the normal control rods.

The vessel contains 8 universal bosses which can accommodate a wide variety of pressure transducers, and 13 bosses for thermocouples. Provision is also made for the installation of a second smaller horizontal through-tube 0.938-in. ID at right angles to and 2.75 in. above the larger tube.

Cooling of the solution is provided by cooling coils of stainless steel tubing 1/4-in. OD with 0.028-in. wall. Ten such coils, each containing 11 ft of tubing, are connected to inlet and outlet manifolds inside the core vessel.

The control rod system was the same as that used for the "A" core tests. The reflector consists of machined AGOT-grade graphite bars stacked to form a 56-in. cube. Adjacent to the core, large blocks of graphite were machined to fit the shape of the core on the inside. Regular 4-in. square bars were stacked around this inner rectangular section to complete the 56-in. cube. The core and reflector are enclosed in the aluminum secondary enclosure.

3. Designation of Core Loadings

During the investigation with the "A" core, two different core loadings were tested. The first was with the core vessel 85% full of fuel solution and will be referred to as Core 1. This represented the normal loading for water boilers that were in use. To investigate the effects of increased fuel volume, and decreased void above the fuel, a series of tests were made with the core 100% full of fuel solution. This is designated as Core 2. With both these core loadings the normal graphite reflector for the "A" core was in place.

During the investigation with the "B" core, three loadings were tested. The first, Core 3, was with 18 l of fuel solution in the core vessel. This gave a fuel depth of 12 in. and a void above the fuel of 17 in. The second loading tested, Core 4, was with 26 l of fuel solution. This provided information on the effects of greater fuel depth and also decreased uranium concentration. For both of these tests the complete graphite reflector was in place.

The third loading tested in the "B" core, Core 5, was with 24 l of fuel solution and no reflector. This bare core investigation was performed to determine the kinetic behavior of aqueous homogeneous reactors without the presence of reflector (delayed) neutrons.

Figure 3 gives a graphical representation of the five core loadings tested. A summary of the reactor characteristics for these five cores is given in Table I.

D. INSTRUMENTATION

The conventional reactor control console and the instrumentation for initiating, detecting, and recording the transient which was used for these tests has been described in detail in an earlier report on the "A" core experiments.¹ A brief description of the type of system used and of the changes or additions made during the "B" core tests will be given here.

All normal reactor operations such as startup, stable power operation and shutdown were controlled and recorded with the normal reactor instrumentation that is installed on all small research reactors. For fast power excursions these conventional instruments are too slow to follow and record the power accurately. Thus a complete set of transient instruments was developed and installed to give an accurate record of reactor power, period, and pressures during large excursions.

Figure 3. Graphical Representation of the Five Cores Tested

For power recording it was discovered early in the tests that Log N systems were too slow to give accurate power records for fast transients. A linear system which recorded the top three decades of the power on three separate channels was developed. To adjust the relative sensitivities of these channels such that they would be in the ratios of 1:10:100, shields were constructed for the uncompensated boron activated ion chambers used in these channels. Shields consisting of a region of water, and then cadmium next to the chamber to decrease the neutron sensitivity, made the ratio of gamma radiation to neutron sensitivity of the chambers large and caused distortion of the power record by holding up the trailing edge of the pulse. An investigation⁵ was carried out, and shields employing cadmium wrapping outside an 8-in. lead shield were found to give the desired decrease in sensitivity without distorting the power trace.

TABLE I
SUMMARY OF KEWB REACTOR CHARACTERISTICS

	Spherical ("A") Core		Cylindrical ("B") Core		
	85% Full 11.45 Liters	100% Full 13.65 Liters	18-Liter Reflected	26-Liter Reflected	24-Liter Unreflected
Critical Mass (gm U ²³⁵)	1680	1240	1450	1290	3900
Total gm U ²³⁵ with Excess Loading	1905	1450	1690	1490	4870
Fuel Concentration (gm U ²³⁵ /Liter)	166	106	94	57	203
H/U ²³⁵ Ratio	152	240	272	450	124
Excess Reactivity (\$)	3.50	3.75	5.80	4.50	3.75
ℓ_o/β (msec)	2.0	2.75	2.60	3.25	1.84
Maximum Peak Reactor Power (Mw)	470	650	2500	1000	4000
Minimum Stable Reactor Period (msec)	2.0	2.0	0.90	2.1	0.56
Energy Released in a Maximum Burst (Mw-sec)	3.1	5.7	10.0	6.1	7.0
Maximum Inertial Pressure Observed (psi)	140	400	335	130	400
Maximum Vessel Acceleration Observed (g's)	-*	-*	130	40	135
Mass Coefficient of Reactivity (\$/gm U ²³⁵)	0.014	0.016	0.016	0.024	0.0035
Water Coefficient of Reactivity (\$/cc)	0.0027	0.0009	0.0006	-0.00035	0.001
Temp. Coefficient of Reactivity at 30°C (\$/°C)	-0.020	-0.041	-0.041	-0.037	-0.053
Gas Production Coefficient (Liters H ₂ -O ₂ /kw-hr)	15.6	18.1	17.3	19.1	15.1

*Core vessel not equipped with accelerometer

Dynamic pressures in the core vessel were detected and recorded with an unbonded strain wire transducer in a balanced bridge network driven by a 20-Kc carrier amplifier. During one phase of the "A" core tests, a special fast pressure recording system was installed to determine the pressures generated from hydrogen-oxygen explosions initiated in the stoichiometric H_2-O_2 gas mixture evolved from the core during operation. A detailed description of the system is given in Reference 1.

All experimental data were recorded with light beam galvanometers in a multichannel, high-speed oscillograph.

Immediate, direct readout of reactor period was obtained from time interval meters which were used in connection with the most sensitive power channels and calibrated such that they were gated on when the output of this channel reached 10 v and gated off when this voltage increased by a factor of e. The time recorded by the time interval meter is then the reactor period. Periods down to 0.60 msec could be measured with an accuracy of ± 0.05 msec.

The "B" core was also equipped with a 500-g accelerometer to measure the accelerations given to the system when the fuel solution was suddenly expelled upward during a large excursion.

Transients were initiated by a large boron carbide rod which operated horizontally in the through tube. This poison rod was pneumatically expelled with a cylinder-piston arrangement operated at up to 120 psi, and was capable of introducing the maximum reactivity, $\$6.25$, in about 20 msec.⁶

The procedure for initiation of a transient was as follows. First, criticality was established with only the four control rods within the core. Then the poison rod was fully inserted. Next, the control rods were raised to a supercritical elevation determined from the control rod calibration curves to give the desired reactivity release or reactor period. Finally, the poison rod was rapidly ejected and the reactor was left in the desired supercritical condition. This method was reliable and accurate; a 1.0 msec period, for example, could always be realized within $\pm 50 \mu\text{sec}$.

E. EXPERIMENTAL RESULTS

1. Burst Phenomenology

The following is a brief description of the physical events of a typical large power burst in the KEWB cores.

Following a large reactivity injection, the reactor responded with a power burst that reached power densities up to 180 Mw/l of fuel solution and maximum fluxes up to $5 \text{ to } 10 \times 10^{16}$ nv. Radiation intensities in the room were sufficient to light the reactor room brilliantly through air ionization, even with the 2-ft graphite reflector in place. Plastic and water-filled objects in the room yielded intense Cerenkov self-images. Argon-41 production from each burst resulted in the reactor room atmosphere being at least ten times the recommended breathing tolerance and the building had to be thoroughly ventilated after each experiment.

For reactivity insertions up to about \$1.25 the only effects of the burst were temperature rise, bubbling, and splashing of the fuel solution. These compensate reactivity and are responsible for the self shutdown of the reactor. When the reactivity insertion was large enough to yield reactor periods of about 20 msec, however, in addition to the above effects, pressures in the reactor core were observed during the shutdown. This inertial pressure, resulting from the rapid production of radiolytic gas, increased with decreasing reactor period. The inertial pressure is the factor which limits the maximum amount of reactivity which may be safely released. In the KEWB reactor, allowing a safety factor of at least four from yield of the core vessel, reactivity releases producing pressures as high as 400 psi were studied. Typically, following the burst, the reactor power level would decay to a level between 50 to 60 kw and then decrease less rapidly due to heating with no coolant flow. For a few special runs the maximum reactivity was installed and the reactor left to its own devices for a period of half an hour. In these cases the reactor power decreased to about 4 to 6 kw in the first 5 min and then maintained this low power for the rest of the test. The reactivity installed was consumed by neutron leakage due to steam bubbles. When full cooling was applied to the core, the reactor behavior was similar, except the final power level at equilibrium was from 50 to 100 kw.

The reactor was found to be stable for all loadings under every condition investigated. That is, no uncontrolled or divergent oscillations were encountered. During the "A" core tests it was discovered that with the core completely filled with fuel solution, chugging occurred when the reactor was operated at steady power levels greater than a few kilowatts. This was probably caused by the limited free surface area at which the liquid and gas could separate. The inability of the gas to escape freely would cause fuel solution to be carried up into the central tube of the overflow chamber, and therefore decrease the reactivity. As the power dropped with a corresponding decrease in gas formation, the fuel would again fall back into the core and the power would rise. The process would repeat every few seconds.

2. Experimental Data

Table I gives a summary of the reactor parameters and maximum transient response of the five core loadings tested. The fuel volumes vary over a range greater than a factor of two, the fuel concentration varies by more than a factor of 3.5, and the reactivity coefficients have similar ranges. The results of the transient tests, in terms of expansion pressure, maximum power, and energy release for a given reactor period have a range of values like the coefficients, volumes, and fuel concentrations. The exception to this is the increased expansion pressures in Core 2 caused by the restriction at the neck of the sphere.

Various initial conditions such as temperature, pressure and power were investigated and their effect on burst magnitude and/or pressure was found to be negligible. The two parameters which governed the kinetic response of the reactors were amount of reactivity insertion and the rate at which it was released.

3. Transient Tests Conducted

Step inputs of reactivity were investigated for releases up to the excess loading for each core as given in Table I. For all cores, the initial fuel temperature was 25°C, initial power level less than 10^{-3} watts, and no coolant was flowing. The core atmosphere was stoichiometric H_2-O_2 at a pressure of about 1 atm abs. Special procedures were followed before each transient to insure that the fuel solution was saturated with H_2-O_2 .

Data were obtained to determine the effect of total reactivity input on the maximum power and pressure. Figures 4 and 5 are artist reproductions of typical transient recordings of the power and pressure for core loadings 3 and 5, respectively.

Figure 4. Typical Transient Recording, Core 3

In Figure 6 the maximum power achieved by the five cores tested following a step input of reactivity is plotted vs the stable period which resulted from the reactivity release. Tabular data for these tests are given in Tables II, III, IV, V, and VI. Figure 6 shows the maximum power achieved for a given stable period is higher for cores with lower fuel concentrations and longer prompt neutron lifetimes.

Figure 5. Typical Transient Recording, Core 5

Some tests with each core were conducted with initial parameters varied. It was found that varying the initial system pressure over the range from 15 in. of Hg abs to 5 atm abs caused only small changes in the maximum power and pressures attained in the burst.

The maximum inertial pressures detected in the cores during these step input transients are shown in Figure 7. No appreciable pressures were generated in any of the cores during transients with stable periods longer than 10 msec.

Figure 6. Maximum Power as a Function of Reactor Period, all Cores

As Figure 7 shows, for a given period, the inertial pressures were greatest in the "A" core where the expansion of the fuel solution was restricted by a decreasing free surface on expansion. Thus, the largest pressures were generated in Core 2, where the free surface area was limited to the 2-in. diam effluent tube at the top of the core. For the "B" core, Cores 3, 4, and 5, the maximum pressures, for a given period, are in the same order as the maximum powers. The maximum pressure observed with any of the 5 loadings was 620 psi.

TABLE II
EXPERIMENTAL DATA - CORE 1

Transient No	Reactor Period (ms)	Peak Power Release (Mw)	Total Energy Release (Mw-sec)	Energy Release to Peak (Mw-sec)	Initial Temperature (°C)	Initial Pressure (cm Hg Abs)	Peak Pressure at Bottom of Core (psi)	Final Temperature (°C)
216	3 01	251	-	-	25 3	45 0	79	58 5
217	2 90	250	-	-	25 1	69 8	52	57 5
218	2 32	318	2 68	1 34	24 7	15 8	91	71 6
219	2 17	319	2 69	-	25 4	44 8	91	66 3
220	2 46	311	-	-	25 1	70 6	97	67 0
221	2 15	380	3 11	1 33	23 7	13 0	131	80 4
222	2 18	364	3 14	1 39	24 7	45 0	107	70 8
223	2 12	381	2 78	1 37	25 1	43 5	112	74 0
224	2 11	378	-	-	25 3	69 9	120	74 0
225	3 97	164	1 86	0 994	25 4	70 1	52	54 5
226	3 88	174	1 87	1 04	23 6	45 3	56	54 5
227	3 77	176	1 88	1 01	25 7	13 8	56	54 0
228	5 97	92 3	1 36	0 861	25 3	15 5	30	47 0
229	6 52	86 0	1 31	0 824	25 1	45 9	17	47 5
230	6 18	96 0	1 44	0 943	25 7	70 9	31	50 3
231	10 8	44 1	1 12	0 764	25 8	70 4	14	46 0
232	11 3	42 0	1 10	0 792	25 8	46 7	16	45 5
233	10 1	46 5	1 07	0 735	25 3	14 2	18	43 5
234	51 5	2 98	0 456	0 317	25 3	14 9	1 4	34 1
235	32 0	6 60	0 761	0 440	25 4	70 4	2 6	48 5
236	28 5	8 00	0 781	0 479	25 4	46 2	3 1	52 3
237	134	0 702	0 605	0 233	26 1	69 8	-	38 5
238	136	0 665	0 552	0 228	25 5	45 9	35	37 0
239	146	0 626	0 459	0 226	25 8	13 6	-	36 6
240	321	0 310	0 427	0 272	25 6	15 0	-	35 6
241	360	0 290	0 544	0 333	27 1	70 0	-	37 8
242	278	0 340	0 537	0 300	25 8	45 7	-	38 2
243	1,250	0 108	-	-	26 1	44 8	-	36 0
244	920	0 131	0 382	0 284	25 1	14 0	-	35 0
245	1,050	0 128	0 638	0 376	25 4	70 7	-	39 0
246	6,120	0 026	0 742	0 380	25 8	70 4	-	38 9
247	6,600	0 026	0 660	0 360	25 4	45 3	-	37 4
248	5,270	0 026	-	-	25 3	13 8	-	35 5
249	980	0 099	0 345	0 191	24 9	5 20	-	32 0
250	1,250	0 122	0 790	0 429	25 8	145	-	32 5
251	48,500	0 002	-	-	24 7	69 8	-	32 3
252	53,300	0 002	-	-	25 3	13 1	-	30 1
253	3,310	0 054	0 842	0 431	25 1	126	-	56 9
254	1,580	0 102	-	-	25 1	137	-	41 0
255	1,430	0 110	0 860	0 445	25 1	138	-	56 0
256	1,620	0 101	0 876	0 435	25 7	137	-	40 5
257	1,450	0 084	0 624	0 269	25 8	12 7	-	36 9

TABLE II (Continued)

Transient No	Reactor Period (ms)	Peak Power Release (Mw)	Total Energy Release (Mw-sec)	Energy Release to Peak (Mw-sec)	Initial Temperature (°C)	Initial Pressure (cm Hg) (Abs)	Peak Pressure at Bottom of Core (psi)	Final Temperature (°C)
258	1,520	0 080	0 615	0 265	25 4	13 4	-	36 6
259	1,450	0 099	-	-	25 4	47 3	-	41 0
260	1,770	0 088	0 673	0 391	25 3	46 7	-	39 0
261	1,370	0 105	-	-	25 3	45 4	-	39 6
262	1,280	0 115	0 789	0 399	25 1	69 8	-	39 6
263	5 10	148	-	-	26 0	69 5	39	57 7
264	2 16	401	-	-	25 3	69 8	99	76 0
265	3 22	247	-	-	25 4	70 0	61	61 5
266	4 66	175	-	-	25 1	70 4	43	35 9
267	8 13	66 5	1 22	0 816	25 9	14 7	21	44 7
268	4 57	148	-	-	25 8	15 3	40	52 6
269	3 18	235	-	-	24 4	15 3	61	94 9
270	2 14	373	2 98	1 34	25 3	14 6	97	66 5
271	3 26	230	-	-	25 3	44 3	59	67 0
272	3 18	230	-	-	26 0	70 1	59	58 5
273	3,280	0 028	0 153	0 121	23 7	15 5	-	26 3
274	22,200	0 004	-	-	25 3	15 4	-	28 2
275	3,130	0 033	-	-	25 8	15 5	-	29 2
276	22,900	0 005	-	-	26 0	14 8	-	29 2
277	19 0	15 1	0 657	0 457	25 3	15 6	-	35 8
278	21,900	0 006	-	-	24 9	69 9	-	30 2
279	2 13	409	3 14	1 37	23 1	15 6	-	68 9
280	1 95	479	-	-	23 5	16 1	-	72 4
281	1 90	498	-	-	23 9	-	-	69 0
282	1 96	465	-	-	22 8	14 9	-	69 4
283	2 08	461	-	-	24 9	15 0	-	68 0
284	2 95	248	-	-	25 4	15 0	-	54 0
285	2 93	251	-	-	25 1	15 2	140	56 0
286	4 99	124	-	-	25 0	15 2	42	47 5
287	7 95	68 5	1 27	0 841	24 4	15 1	16	-
288	7 78	73 0	1 34	0 877	26 7	70 3	6 5	50 4
289	5 03	128	-	-	25 9	69 5	30	50 6
290	2 98	268	-	-	25 4	70 5	104	60 5
291	2 05	450	-	-	25 9	70 2	220	65 5
292	17 1	21 3	0 982	0 701	26 5	70 3	-	42 5
293	1,950	0.085	0 614	0 392	26 3	70 1	-	37 4
294	4,240	0.036	-	-	26 0	69 9	-	37 4
295	2 04	-	-	-	25 7	70 2	274	20 8
296	17,600	0.0073	-	-	25 6	70 0	-	33 1
297	17,300	0.0057	-	-	25 8	14 8	-	30 4
298	10,900	0.011	-	-	24 9	15 2	-	31 4
300	33 2	6 64	0 715	0 437	20 0	65 9	-	-
301	35 2	6 02	-	-	24 9	65 9	-	-

TABLE III
EXPERIMENTAL DATA - CORE 2

Trans- ient No	Reactor Period (ms)	Peak Power Release (Mw)	Total Energy Release (Mw-sec)	Energy Release to Peak (Mw-sec)	Initial Temper- ature (°C)	Initial Pressure (cm Hg) (Abs)	Peak Pressure at Bottom of Core (psi)	Final Temper- ature (°C)
171	22 8	12 2	-	-	24 8	14 6	-	-
172	20 5	12 4	0 74	0 45	25 7	14 7	5 6	40 6
173	24 3	11 9	0 88	0 53	25 7	70 1	4 1	39 6
174	11 4	43 1	1 33	0 89	25 7	70 2	29 6	43 6
175	7 72	79 2	-	-	26 1	70 2	-	50 5
176	5 25	137	2 13	1 23	25 5	70 1	93	54 1
177	4 09	200	2 59	1 39	25 6	70 2	135	61 5
178	3 50	226	-	-	24 6	70 1	154	25 8
179	3 33	282	-	-	26 5	70 0	195	68 4
180	2 92	331	-	-	27 5	70 4	186	71 6
181	2 68	366	3 89	1 70	26 6	71 5	214	84 5
182	2 12	621	-	-	26 1	69 0	435	91 9
183	1 82	773	6 38	2 77	26 5	68 0	620	104
184	3 32	280	-	-	25 8	68 7	181	65 4
185	9 60	48 9	-	-	22 5	15 2	28	38 3
186	5 03	138	2 02	1 14	24 4	14 2	83	48 8
187	3 73	204	-	-	25 4	13 7	120	56 0
188	2 92	341	-	-	-	14 7	203	-
189	2 54	441	-	-	25 0	15 2	248	74 0
190	2 12	592	-	-	24 4	14 3	376	88 2
191	1 98	682	5 82	2 49	26 5	14 6	442	91 7
192	2 38	492	-	-	25 8	67 4	282	82 0
193	109	0 963	-	-	26 4	16 5	-	30 5
194	1,030	0 106	-	-	27 0	16 8	-	29 2
195	10,100	0 012	-	-	23 2	16 9	-	32 9
196	105,000	0 001	-	-	21 9	16 7	-	26 7
197	297	0 267	-	-	25 3	16 4	-	29 0
198	75,100	0 001	-	-	25 7	70 9	-	28 2
199	8,490	0 012	-	-	26 4	71 2	-	31 0
200	30,600	0 003	-	-	26 2	70 3	-	29 1
201	1,010	0 090	0 27	0 17	26 2	70 6	-	29 7
202	300	0 248	0 44	0 24	25 6	70 2	-	32 0
203	117	0 630	0 45	0 17	26 1	70 4	-	32 9
204	152	0 448	-	-	26 1	71 0	-	33 0
205	9 55	48 2	1 40	-	-	-	50	44 6
206	891	0 103	0 282	-	-	-	-	29 6
207	9,500	0 011	-	-	-	-	-	29 1
208	110	0 705	0 348	-	-	-	-	30 2
209	2 38	525	5 33	-	-	-	22	78 6
210	10 1	43 2	1 23	0 755	25 4	4 6	12	54 3
211	1 97	707	6 84	2 93	23 7	14 8	420	85 5
212	53 8	2 14	0 337	0 213	25 5	14 6	-	30 5
213	3,020	0 031	0 176	0 139	25 6	14 8	-	28 2
214	16 4	18 2	0 839	0 509	25 6	14 8	-	36 0
215	26 3	8 53	0 592	0 403	24 9	15 6	-	33 0

TABLE IV
EXPERIMENTAL DATA - CORE 3

Transient No.	Reactor Period (ms)	Peak Power Release (Mw)	Total Energy Release (Mw-sec)	Energy Release to Peak (Mw-sec)	Initial Temperature (°C)	Initial Pressure (cm Hg) (Abs)	Peak Pressure at Bottom of Core (psi)	Peak Acceleration (g)	Final Temperature (°C)
1001	533	0.208	0.491	0.295	24.7	69.1	-	-	34.4
1002	228	0.370	0.512	0.262	26.1	69.6	-	-	33.8
1003	76.4	1.34	0.585	0.220	24.7	70.0	-	-	33.4
1004	39.6	4.26	0.754	0.344	22.9	69.7	-	-	33.6
1005	16.5	21.5	1.12	0.708	24.7	69.5	1.8	0.1	40.6
1006	9.57	57.3	-	-	25.8	70.2	9.1	0.8	46.6
1007	4.60	171	-	-	20.9	69.5	31	1.8	51.1
1008	5.15	140	-	-	26.1	68.7	27	0.05	52.1
1009	3.77	218	-	-	25.8	69.8	43	1	62.9
1010	2.46	401	2.80	1.54	26.8	69.0	80.7	11	65.1
1011	1.85	643	-	-	26.2	65.6	123	22	74.7
1012	1.54	880	-	-	26.9	70.3	157	31	84.1
1013	5.07	151	2.03	-	23.5	61.2	29	3	-
1020	1.30	1250	6.39	-	25.6	69.2	197	43	92.8
1021	1.15	1550	7.16	3.29	25.9	62.2	244	43	103
1022	0.935	2220	9.02	-	25.8	69.3	354	46	116
1028	0.915	2300	-	-	25.1	68.4	309	120	119
1029	2.36	502	3.41	1.83	25.4	69.0	98	22	64.1
1030	1.48	1050	6.12	2.47	23.4	69.1	154	53	84.8
1031	0.920	2360	10.4	4.00	25.3	69.1	340	110	113
1032	1.03	1830	-	3.52	23.1	67.3	259	94	83.5
1033	2.12	331	2.61	1.47	25.2	60.2	63	15	44.0
1037	4.98	170	-	-	27.6	70.3	29	2	54.3
1038	2.50	449	3.26	1.77	25.1	70.1	75	9	69.1
1039	7.03	71.3	-	-	72.0	69.7	15	2	84.1
1040	3.69	179	1.72	1.01	72.1	69.4	37	6	91.2
1041	3.18	218	-	-	73.5	70.0	45	9	94.0
1042	1.71	584	-	-	87.0	67.9	108	30	115
1044	211	0.427	0.453	0.243	24.6	70.5	-	-	28.9
1045	309	0.321	4.98	0.269	26.2	69.8	-	-	33.3
1046	86.7	1.28	0.561	0.231	25.8	68.9	-	-	33.7
1047	123	0.783	0.549	0.219	25.5	68.6	-	-	33.5
1049	32.4	6.79	0.797	0.432	24.6	68.1	-	-	36.2
1050	21.1	15.2	1.10	0.619	25.6	69.1	-	-	40.6
1051	1.14	1690	8.05	3.64	24.1	69.3	234	1	107
1052	16.4	23.0	-	-	24.5	70.0	1.3	-	39.3
1053	1.20	1410	-	-	25.6	70.0	212	68	101
1054	1.26	1310	6.49	-	26.7	69.3	209	67	95.8
1055	1.31	1210	-	-	23.1	69.0	194	60	92.3
1056	1.43	1030	-	-	25.0	68.3	174	52	87.7
1057	1.49	904	-	-	23.8	69.4	162	48	83.0
1058	25.8	9.85	0.700	0.452	25.6	69.5	-	-	35.3
1059	1.23	1370	6.47	-	25.6	15.0	188	68	93.2
1060	1.29	1360	6.63	3.09	25.7	70.0	214	67	99.7
1061	14.2	30.2	1.18	0.780	23.3	68.8	2.7	-	38.8
1062	7.2	98.5	1.66	1.13	22.6	69.0	17	2	43.9
1063	3.43	291	2.55	-	24.2	67.2	54	11	53.6
1064	10.3	52.1	-	-	26.0	67.4	7.3	1	44.0

TABLE IV (Continued)

Transient No.	Reactor Period (ms)	Peak Power Release (Mw)	Total Energy Release (Mw-sec)	Energy Release to Peak (Mw-sec)	Initial Temperature (°C)	Initial Pressure (cm Hg Abs)	Peak Pressure at Bottom of Core (psi)	Peak Acceleration (g)	Final Temperature (°C)
1065	10.3	51.1	1.30	0.926	25.0	15.0	9.0	1	43.0
1066	131	0.679	0.397	0.210	19.0	15.1	-	-	24.6
1067	143	0.613	0.480	0.212	23.8	69.9	-	-	31.0
1068	20.2	15.1	-	-	24.7	68.9	-	-	39.7
1069	672	0.182	0.531	0.338	25.1	68.5	-	-	33.5
1071	5.21	150	1.85	-	15.0	15.0	30	5	47.2
1073	712	0.156	0.216	0.170	15.3	14.9	-	-	28.0
1078	373	0.247	2.55	1.97	14.9	14.4	-	-	-
1080	425	0.221	-	0.201	26.2	69.3	-	-	29.5
1125	10.8	47.2	1.35	0.966	24.7	67.5	-	-	-
1126	22.4	14.6	1.12	0.638	24.5	67.6	-	-	37.6
1127	38.1	3.93	0.655	0.300	22.2	66.4	-	-	33.3
1131	2.00	659	3.62	2.11	24.1	64.5	-	-	70.9
1132	1.48	678	-	-	-	-	-	-	68.4
1133	0.980	2360	-	-	23.7	58.2	-	-	96.7
1134	1.93	528	-	-	25.8	68.1	-	-	72.1
1135	1.80	701	-	-	26.3	68.6	-	-	73.3
1136	1.83	687	-	-	26.4	70.0	-	-	72.2
1137	1.87	710	-	-	26.4	67.8	-	-	73.3
1140	1.80	722	3.94	2.25	25.6	66.2	-	-	71.5
1141	0.95	1940	7.93	3.90	25.0	69.0	-	-	113
1144	66.5	1.80	0.652	0.257	25.6	67.4	-	-	34.8
1145	63.2	2.10	0.769	0.281	20.0	70.5	-	-	32.1
1146	21.4	11.9	1.09	0.570	36.5	69.5	-	-	41.3
1148	8.42	75.3	1.59	1.15	22.1	70.3	12	-	42.8
1149	4.68	183	2.03	1.34	36.2	68.2	35	6	52.6
1150	3.85	251	2.09	1.49	24.4	71.5	48	8	54.1
1151	2.91	390	2.88	1.75	23.3	71.0	67	15	58.3
1153	1.42	1190	5.54	2.97	27.7	109	166	50	90.4
1154	5.23	165	2.10	1.41	26.0	84.1	25	5	51.4
1156	1.04	2330	-	-	24.9	40	-	-	103.3
1157	1.06	2100	8	4.13	23.6	60.1	251	82	104
1160	1.10	2100	8.18	4.12	26.9	70.0	254	83	102
1162	3.05	351	2.77	1.69	27.1	68.1	-	13	56.9
1164	1.89	660	-	-	25.0	69.9	-	-	70.3
1166	1.41	1320	5.73	3.10	25.0	70.0	178	18	-
1167	1.31	1490	6.29	3.32	26.3	68.4	199	15	90.6
1169	2.12	681	-	-	25.4	69.7	110	29	74.5
1170	1.2	1750	6.81	3.58	26.0	65.5	249	71	97.4
1171	1.06	2360	8.31	4.34	25.2	70.5	283	94	110
1172	1000	0.136	0.426	0.276	26.4	68.0	-	-	31.2
1173	769	0.173	0.477	0.324	25.3	69.7	-	-	31.6
1184	1.94	774	4.22	2.42	22.8	69.0	110	29	71.3
1185	5.15	172	2.07	1.41	23.0	68.0	13	5	48.7
1186	2.97	403	3.05	1.87	22.0	68.4	52	14	58.0
1188	2.98	362	2.74	1.68	23.6	66.5	67	13	58.0
1190	1.02	2120	7.57	3.88	25.8	69.3	276	95	112

TABLE V
EXPERIMENTAL DATA - CORE 4

Transient No.	Reactor Period (ms)	Peak Power Release (Mw)	Total Energy Release (Mw-sec)	Energy Release to Peak (Mw-sec)	Initial Temperature (°C)	Initial Pressure (cm Hg) (Abs)	Peak Pressure at Bottom of Core (psi)	Peak Acceleration (g)	Final Temperature (°C)
2001	208	0.521	0.750	0.306	24.4	69	-	-	34.0
2002	90.0	1.54	0.904	0.291	25.5	69.4	-	-	33.0
2003	61.0	2.90	1.05	0.357	24.0	70.1	-	-	33.2
2004	37.4	7.65	1.28	0.543	26.6	69.5	-	-	36.9
2005	22.0	19.0	1.51	0.826	24.5	69.0	-	-	37.6
2006	15.0	38.6	1.68	1.16	26.8	69.2	3.0	0.4	42.7
2007	10.5	81.9	1.97	1.40	27.0	69.3	9.6	1	42.7
2008	7.30	131	2.25	1.55	24.0	68.8	19	3	42.7
2009	5.60	195	2.60	1.73	26.6	69.5	29	4	47.0
2010	4.40	298	3.12	1.96	25.6	68.0	42	7	50.7
2011	3.20	468	3.90	2.35	25.8	67.6	66	12	57.1
2112	2.40	850	5.42	3.08	24.4	69.5	117	25	67.8
2013	2.90	549	4.15	2.48	24.8	68.1	75	14	58.9
2015	2.30	810	5.24	3.03	24.9	67.0	112	22	66.1
2016	958	0.147	0.468	0.297	24.0	68.1	-	-	30.1
2017	3.25	455	3.78	2.29	26.5	67.5	64	11	56.9
2022	20.3	23.2	1.52	0.914	22.8	69.2	-	-	35.0
2023	9.60	77.9	1.80	1.29	22.8	71.5	12	2	40.1
2025	105	1.29	0.799	0.297	22.7	70.6	-	-	29.6
2026	37.8	7.20	1.05	0.552	24.0	70.0	-	-	32.8
2028	22.2	20.0	1.47	0.872	26.5	71.4	-	-	38.1
2029	10.2	86.8	2.11	1.51	23.9	71.5	8.0	1	41.1
2035	21.6	21.6	1.50	0.822	22.1	69.2	-	-	35.0
2036	10.0	89.1	2.08	1.48	22.6	71.5	11	1	40.2
2037	5.10	251	2.93	1.92	23.0	70.2	98	5	47.1
2038	2.60	738	5.05	2.95	23.0	70.6	240	77	60.8
2039	2.70	713	-	-	22.5	70.5	81	15	60.5
2040	101	1.36	0.816	2.99	22.3	68.7	-	-	29.1
2041	9.70	91.1	2.16	1.53	22.6	71.0	8.0	1	39.1
2042	5.10	257	3.06	2.01	18.6	71.4	33	5	44.3
2043	2.60	736	5.13	2.99	23.0	71.0	86	17	60.5
2044	8.50	119	2.35	1.63	20.2	71.5	13	2	39.0
2045	9.60	95.1	2.21	1.58	22.1	48.0	6.0	1	38.5
2046	4.00	369	3.60	2.26	21.7	60.2	48	7	49.7
2047	2.40	848	5.56	3.20	23.1	72.0	97	19	64.1
2055	9.80	84.1	1.96	1.39	30.2	72.0	7.5	1	43.3
2056	9.70	94.2	2.19	1.58	23.5	71.7	9.0	1	40.6
2057	2.35	870	5.73	3.27	22.6	72.4	103	21	65.1
2062	2.5	839	5.67	3.32	21.2	70.8	97	18	61.8
2063	9.6	91.7	2.11	1.50	21.9	71.6	8.0	1	39.5
2064	7.6	132	2.34	1.63	23.9	71.4	10	2	43.3
2065	20.1	22.4	1.52	0.872	23.7	71.1	-	-	37.7
2066	2.5	798	5.29	3.11	23.7	71.5	19	20	63.0
2072	2.05	1060	6.09	3.39	22.7	71.3	138	28	71.9

TABLE VI
EXPERIMENTAL DATA - CORE 5

Trans- ient No.	Reactor Period (ms)	Peak Power Release (Mw)	Total Energy Release (Mw-sec)	Energy Release to Peak (Mw-sec)	Initial Temper- ature (°C)	Initial Pressure (cm Hg Abs)	Peak Pressure at Bottom of Core (psi)	Peak Acceler- ation (g)	Final Temper- ature (°C)
3014	0.930	1570	4.12	2.54	24.3	38	160	40	61.1
3015	0.860	1930	4.53	2.77	21.0	62.6	190	50	61.7
3016	0.740	2380	5.02	2.96	25.1	71.8	240	60	71.5
3017	0.670	2980	5.73	3.42	26.1	70.4	250	100	78.5
3018	0.570	4080	7.04	4.16	25.4	73.1	390	135	87.6
3024	0.560	4090	7.00	4.10	26.8	70.5	400	120	89.2
3025	0.830	1690	-	-	25.9	69.5	200	50	65.6
3026	1.56	518	2.02	1.34	26.9	70.9	65	20	49.8
3027	0.810	1940	4.42	2.68	25.9	71.3	195	60	66.8
3028	223	0.302	0.368	0.183	26.5	71.5	-	-	29.8
3029	105	0.460	0.382	0.173	26.6	70.7	-	-	30.0
3030	34.9	1.25	0.446	0.115	25.1	71.3	-	-	30.6
3031	108	0.463	0.315	0.197	26.1	71.3	-	-	31.0
3032	34.6	1.49	0.402	0.112	25.8	71.4	-	-	30.5
3033	31.6	1.70	0.450	0.116	26.8	71.5	-	-	31.3
3034	12.6	9.02	0.540	0.234	25.4	71.4	-	-	32.4
3035	7.80	24.6	0.791	0.378	25.0	70.8	-	-	35.1
3036	5.10	53.4	1.01	0.548	22.7	72.9	-	-	48.3
3037	3.00	134	1.13	0.763	30.3	70.1	27	4	45.3
3038	1.99	339	1.75	1.20	26.2	70.4	42	100	45.8
3039	1.76	400	1.86	1.26	25.8	69.5	53	6	47.6
3040	1.61	466	1.99	1.33	26.7	71.5	67	8	48.3
3041	1.45	611	2.28	1.50	24.8	71.2	85	5	49.4
3042	1.48	605	2.09	1.39	25.8	70.2	75	20	49.3
3043	1.26	770	2.56	1.67	26.5	69.3	100	20	53.1
3044	1.02	1200	2.56	1.62	24.5	69.3	142	30	58.9
3045	0.950	1200	3.36	2.11	25.9	68.4	144	40	60.5
3046	506	0.216	-	-	24.6	848	-	-	-
3047	260	0.309	1.04	0.307	24.5	848	-	-	-
3048	87.9	0.541	-	-	22.3	848	-	-	-
3049	121	0.455	0.729	0.213	25.2	848	-	-	-
3050	27.0	2.19	0.586	0.130	25.5	848	-	-	-
3051	17.5	4.86	0.467	0.174	24.9	848	-	-	-
3052	5.43	47.7	1.07	0.518	25.7	848	-	-	-

Figure 7. Maximum Inertial Pressure as a Function of Reactor Period, all Cores

The effect of varying the initial temperature of the fuel solution was investigated for several different reactor periods and two different initial core pressures. The maximum power decreased with increased temperature in all cases investigated. Tests made with low initial pressures showed a sharp decrease in maximum power as the initial temperature neared the solution boiling point at that pressure. Figure 8 shows typical data for the "A" and "B" cores.

The power traces were integrated to obtain values of the energy release in the power pulse. Figure 9 shows the variation of total energy released in the

Figure 8. Effects of Initial Core Temperature on Peak Power, Cores 1 and 2

Figure 9. Energy Release as a Function of Reactor Period, all Cores

burst as a function of the reactor period. Similarly, Figure 10 shows the energy release to peak power. As would be expected, the energy released at a given period in the various cores tested is in the same order as the maximum powers.

Figure 10. Energy Release to Peak Power as a Function of Reactor Period, all Cores

To determine the effect of reactor power on the transient response of the reactor, a series of step reactivity input transients were run with Core 1 at initial stable powers of 1, 10, and 20 kw. The initial fuel temperature for this series of runs was set at 40°C, since the cooling system could not maintain a 25°C fuel temperature at 20 kw stable power. Figure 11 presents the results of these excursions. In this graph the solid line labeled $T_O = 25^\circ\text{C}$ is the result of step inputs with initial power less than 1 mw and initial fuel temperature of 25°C. The dash-line labeled $T_O = 40^\circ\text{C}$ represents this step input data normalized to data with an initial temperature of 40°C to compare with the higher initial power tests. For tests with initial power below one mw, the reactor would rise on a stable period through as much as five or six decades of power before the onset of shutdown during a 1 msec transient. At the higher starting powers this time was reduced sharply, so that, at initial power levels in the kilowatt region, there existed no fixed period by which the burst could be categorized. Under these conditions, present also during ramp reactivity injections, the

Figure 11. Effect of Initial Power on Peak Power, Core 1

minimum period experienced by the reactor was taken as the index of the kinetic response. Bursts resulting from step reactivity inputs which had stable periods equal to the minimum periods observed with the high initial power and ramp input tests were found to have similar maximum powers and burst shape.

A series of experiments in Cores 1 and 2 was initiated in which the reactivity input was a linearly increasing function of time. Reactivity ramps were obtained by a programmed control rod withdrawal plan giving ramp rates from \$0.04 to \$0.16/sec and total reactivity inputs up to \$6.25.

In all cases a power transient occurred early in the ramp input and then the power fell to a low level and slowly increased with long period, low amplitude oscillations until the rod withdrawal was stopped. After the reactivity insertion was terminated, the power level slowly decreased as the fuel temperature increased.

The magnitude of the initial power peak was dependent on ramp input rate, and the maximum power level after the burst on the total reactivity input. The maximum power achieved during the initial burst was 3 Mw for a ramp corresponding to the withdrawal of all four control rods simultaneously. Maximum post burst power was 50 to 60 kw for a total reactivity input of \$6.25.

During the experiments on the "B" core, a 500-g accelerometer was attached to the core vessel. The core was suspended from the upper face of the aluminum box which enclosed the graphite reflector. The face was reinforced with aluminum ribs, but even so, vertical motion up to 0.25 in. was experienced by the core vessel. This motion resulted from the large, momentarily uncompensated downward force ($\sim 35,000$ lb) which existed due to the rapid inertial pressure rise in the fuel solution. Accelerations of up to 130 g were recorded during the largest power excursions. These accelerations made it impossible for the control rod magnets to support their individual loads and all rods separated from the magnets during bursts with periods less than 3 msec during the "B" core tests. The release time of the rods was such that no reactivity perturbation occurred during that portion of the power trace upon which analytical calculations were based. No damage was caused to the core vessel as a result of the large accelerations but a number of minor problems arose in the limit switches and control drive mechanism due to the high accelerations which were transmitted to these components from the core vessel. A 6-in. steel, floor-to-ceiling I-beam structure was constructed to minimize the damage to these components attached to the core vessel.

F. SUMMARY

The data generated in the KEWB program show that small aqueous homogeneous reactors are extremely safe. The release of sufficient reactivity to place such a reactor on a 20-msec period excursions produces no hazardous effects: the reactor shuts itself down with only a small temperature rise. At elevated steady state operating temperatures the system reactivity is reduced so that large excursions are not possible. Radiolytic gas recombination explosions merit consideration in the design and operation of aqueous homogeneous reactors, but do not represent a serious threat to their safe operation.^{1,10}

The release of sufficient reactivity to produce reactor periods appreciably less than 20 msec results in the generation of an inertial pressure in the

reactor as a product of the shutdown process. The KEWB data define the pressure for 12-in. spherical and cylindrical cores with fuel solution volumes from 11.45 to 26 ℓ , U^{235} concentrations from 57 to 203 gm/ ℓ with and without graphite reflector. The minimum period transient produced in the program, 0.56 msec, was reached with the unreflected cylindrical core. The peak power was about 4000 Mw and peak inertial pressure about 400 psi. This allowed a safety factor of over four from the calculated yield pressure of the 0.25-in. thick core vessel. A burst of this size would be beyond the capabilities of a reactor of this type unless specifically designed to contain a large loading of excess reactivity and means to release this reactivity much faster than possible with control rod drives. The data thus go beyond the maximum expected application. The largest pressures for a given reactor period were found in the full spherical core because of its limited free surface area. Operation with limited free surface area is thus possible with increased inertial pressures probable in a fast transient, but should be easy to avoid in design. The data presented in Figures 6 to 10 can be used as a guide for aqueous homogeneous reactor safety evaluations. Means for the computation of characteristics of widely divergent designs will be presented in Section III of this report.

Ramp input of reactivity, such as would result from the accidental withdrawal of all control rods to their all-out position, yield comparatively mild power bursts. In these, the reactor is capable of self-control such that power levels in the kilowatt region will be maintained for long periods even though the control rods remain withdrawn.

The unreflected "B" core tests, Core 5, gave the shortest periods and highest peak powers for a given maximum inertial pressure generated. This is basically due to the shorter neutron lifetime for this loading. With this shorter lifetime short period transients are initiated with smaller reactivity releases and thus less compensation is necessary for power turn around and shutdown. Lower pressures are generated for a given peak power since the pulse is much narrower and the total energy release is smaller.

III. A MODEL OF AQUEOUS HOMOGENEOUS REACTOR KINETICS – M. S. DUNENFELD

A. INTRODUCTION

It is of particular concern, in the design of nuclear reactors, to anticipate possible realms of hazardous or unsafe operation. One of the major areas of this concern is evaluation of the consequences of accidental assembly of supercritical nuclear masses, or, in the case of a reactor, accidental release of large amounts of excess reactivity. The KEWB program, initiated primarily to investigate such hazards in connection with a specific aqueous homogeneous reactor design,⁷ has provided ample data to define the kinetic behavior of that design and similar 12-in. aqueous homogeneous reactor configurations. These data have been presented in Section II of this report and in Part I of the "A" core report.¹

The object of analysis of the KEWB data has been to gain sufficient understanding of inherent reactor dynamics to permit construction of a kinetics model applicable to any aqueous homogeneous reactor. This would permit computation of characteristics of designs differing materially in configuration from the KEWB reactor, hopefully with greater assurance of accuracy than extrapolation of existing data. The effort has met with considerable success, and in this section a remarkably accurate, simple model of aqueous homogeneous reactor kinetics will be presented. Comparisons of bursts predicted by the model will be made with data from four core loadings studied in the KEWB program. Instructions for use of the model in application to hazards evaluations of aqueous homogeneous reactors in general will also be provided.

B. KINETICS MODEL FOR AQUEOUS HOMOGENEOUS REACTORS

For convenience, the kinetics model and nomenclature will be presented here. Discussion of parameters and justification of the shutdown and neutron delay mechanisms will follow. The bare equivalent, space independent kinetics equations are

$$\frac{dN(t)}{dt} = \frac{C}{\ell_o} \left\{ [R(t) - 1] N(t) + \sum_{i=1}^7 f_i W_i(t) \right\} \quad \dots (1)$$

$$\frac{dW_i(t)}{dt} = \lambda_i [N(t) - W_i(t)] \quad \dots (2)$$

The applicable reactivity feedback equation is²

$$R(t) = R_o + (a + bT_o)T(t) + \frac{b}{2}T(t)^2 + DV(t) \quad \dots (3)$$

where

$$R_o = \frac{\omega_o}{C} \left(\ell_o + \frac{C_7}{\omega_o + \lambda_7} \right) + \sum_{i=1}^6 \frac{f_i \omega_o}{\omega_o + \lambda_i} \quad \dots (4)$$

$$T(t) = kE(t) = k \int_0^t N(t') dt' \quad \dots (5)$$

and³

$$V(t) = 0, \quad E(t) < E_o \quad \dots (6)$$

$$V(\tau) = \frac{dA}{\delta h} \int_0^\tau (\tau - \tau') P(\tau') d\tau', \quad E(t) \geq E_o \quad \dots (7)$$

$$E_o = \frac{V_{core} C_c \bar{\phi}_p}{G_{H_2}} \quad \dots (8)$$

$$P(\tau) = \frac{gN_o}{V_{core}} \left(1 - e^{-\omega_o t} \right), \quad 0 \leq t \leq 1.5/\omega_o \quad \dots (9)$$

$$P(\tau) = \frac{gN_o}{V_{core}} \left(1 - e^{-1.5} \right) e^{-\tau/\tau_1}, \quad \tau > 1.5/\omega_o \quad \dots (10)$$

where

t = time (sec)

$N(t)$ = instantaneous reactor power (Mw)

C = effective delayed neutron fraction

ℓ_o = core effective neutron lifetime (sec)

$R(t)$ = reactivity (\$)

f_i = relative yield of the i^{th} delayed neutron precursor group
 $W_i(t)$ = relative concentration of the i^{th} delayed neutron precursor group (Mw)
 λ_i = decay constant of the i^{th} delayed neutron precursor group (sec^{-1})
 R_o = step input reactivity (\$)
 a = linear element of the temperature coefficient of reactivity ($\$/^\circ\text{C}$)
 b = quadratic element of the temperature coefficient of reactivity ($\$/^\circ\text{C}^2$)
 $T(t)$ = temperature increase of the fuel solution during the transient ($^\circ\text{C}$)
 T_o = initial temperature of fuel solution ($^\circ\text{C}$)
 D = void coefficient of reactivity ($\$/\text{cm}^3$)
 $V(t)$ = radiolytic gas void volume (cm^3)
 ω_o = reciprocal stable reactor period (sec^{-1})
 C_7 = effective fractional abundance of reflector delayed neutrons
 λ_7 = reciprocal mean delay time of reflector delayed neutrons (sec^{-1})
 k = reciprocal heat capacity ($^\circ\text{C}/\text{Mj}$)
 $E(t)$ = energy release (Mj)
 E_o = threshold energy release necessary for void formation (Mj)
 τ = time with origin at E_o (sec)
 d = pressure unit conversion constant ($6.9 \times 10^4 \text{ gm}/\text{cm sec}^2 \text{ psi}$)
 V_{core} = fuel solution volume (cm^3)
 $\bar{\phi}_p$ = ratio of average to peak flux
 C_c = critical gas concentration in fuel solution ($\text{gm H}_2/\text{gm solution}$)
 G_{H_2} = hydrogen formation constant per unit energy release (gm/Mj)
 A = cross sectional area of core (cm^2)
 δ = fuel solution density (gm/cm^3)
 h = height of the fuel solution (cm)
 $P(\tau)$ = inertial pressure in fuel solution (at bottom of core)(psi)

g = power-density-void coefficient (psi-cm³/Mw)

N_o = reactor power at time E_o is reached (Mw)

τ_1 = inertial pressure relaxation period (sec)

The units given above have been commonly used in the KEWB program, and are presented as a consistent set. They will be employed throughout the remainder of this report, with the exception of reactivity units, which will be restricted to six group dollars; that is, dollars referred to the six groups of fission product precursor delayed neutrons.

Neutrons which are thermalized in the reflector leak back to the reactor core with a mean delay time which is long compared to the core lifetime. These are treated as a seventh delayed neutron group above, hence the appearance of the terms C_7 and λ_7 .

Reactivity in dollars is defined by

$$R \equiv \frac{k_{ex}}{C} \quad \dots (11)$$

where k_{ex} is the excess multiplication factor. The dollar unit changes materially when reflector thermalized neutrons are considered as delayed neutrons. This is undesirable in comparison of different core loadings and reactors. It is also undesirable because the dollar unit appears to have become associated only with the six groups of fission product delayed neutrons by workers in the nuclear field. Therefore reactivity and reactivity coefficients will be quoted in six group dollars in the following discussions. In the above kinetics model, and when computations are performed, however, the units must be seven group dollars when the reflector delayed neutrons are included.

C. DISCUSSION OF REACTOR PARAMETERS AND EXPERIMENTAL VALUES

We are particularly interested in employing the above kinetics model to compute $N(t)$, $E(t)$, $T(t)$, and $P(t)$. In doing so we will also compute $R(t)$, $W_i(t)$, $V(t)$, E_o , R_o (or ω_o), and N_o . The remaining parameters, which must be specified, fall into four categories:

- 1) Initial conditions: ω_o (or R_o), T_o , $N(O)$, and as tacitly assumed above $V(O) = O$.
- 2) Simple system parameters, easily calculated from system dimensions or available in the literature: f_i , λ_i , k , A , δ , h , V_{core} , and G_{H_2} .
- 3) Nuclear system parameters. These are also calculable, but with considerably more difficulty than any of the parameters in the preceding category. In application of the model to the KEWB reactor, the values employed for these parameters will be (mainly) those determined experimentally: C , ℓ_o , a , b , D , C_7 , λ_7 , and $\bar{\phi}_p$.
- 4) Void model parameters: d , C_c , ϵ , g , and τ_1 .

The parameters in categories 1 and 2 should require no further explanation, although some comment about the f_i being regarded as handbook constants is in order. Fission product delayed neutrons are released to the system with considerably lower energy than prompt fission neutrons. Therefore, the probability of their being thermalized is greater than for prompt neutrons. This makes their effective abundance, C_i , greater than their absolute abundance, β_i .

The f_i are defined by

$$f_i \equiv \frac{\beta_i}{\sum_{i=1}^6 \beta_i} \quad \dots (12)$$

and

$$f_{i\text{-effective}} = \frac{C_i}{\sum_{i=1}^6 C_i} = \frac{p_i \beta_i}{\sum_{i=1}^6 p_i \beta_i} \quad \dots (13)$$

where p_i are the relative probabilities of thermalization for the delayed neutrons.

Since the energy spread of the delayed neutrons is not large, for thermal reactors there is little difference in the p_i . Therefore, with very little error,

$$f_{i\text{-effective}} = \frac{p_i \beta_i}{\sum_{i=1}^6 p_i \beta_i} = \frac{\beta_i}{\sum_{i=1}^6 \beta_i} = f_i. \quad \dots (14)$$

We are thus justified in using the f_i in our equations and calculations (rather than $f_{i \text{ eff}}$).

1. Delayed Neutron Fraction, Core Lifetime, and Reflector Delayed Neutrons

The quantities C , the effective delayed neutron fraction; ℓ_o , the core lifetime; C_7 , the reflector delayed neutron abundance; and λ_7 , the reciprocal mean reflector thermalized neutron delay time, are interrelated in all experiments performed in the KEWB program. There are no independent measurements of any of these quantities. The effective abundance of the fission product precursor delayed neutrons has been assumed to be 0.008 for the KEWB reactors. Nuclear calculations which have been performed have been between this value and 0.0075. We must avoid the temptation of becoming overly involved

with attempts to calculate or measure $\sum_{i=1}^6 C_i$ more precisely. This is because

we have assumed a simple model to be applicable, namely one in which C and, more particularly, the related quantity ℓ_o are invariant. Since we are dealing with reactors which shut themselves down, it is clear that the neutron leakage changes materially throughout the course of a transient. Present technology does not permit experimental determination of the effect this will have on C or ℓ_o . In fact, even static measurements are involved and uncertain. Likewise the core lifetime should be a function of excess reactivity; that is, as the control rods are removed progressively further from the fuel solution to yield shorter period bursts, the system absorption should decrease.* We thus see

there are several sources of uncertainty in the value of $\sum_{i=1}^6 C_i$ (and ℓ_o , as well as

*Under the assumption that this will increase the core lifetime somewhat in proportion to the excess reactivity, k_{eff} , correction has been attempted (while providing simplicity) by omission of the latter term from the inhour equation.

other parameters). We hope to illustrate that the uncertainties are all second order effects by demonstrating close agreement between the predictions of the model and experimental data.

Explicit values for C and ℓ_0 are not required for the six group kinetics model if C/ℓ_0 is known. The latter is usually determined from pile oscillator experiments. In the case of the graphite reflected aqueous homogeneous reactor, where reflector thermalized neutrons may be treated as a seventh delayed neutron group, this simplification would not be appropriate until the reactor period became much shorter than the seventh group delay time. This is at a shorter period than such reactors have been operated. We are therefore faced with the problem of determination of the reflector delayed neutron abundance and delay time from data taken at reactor periods of the same order of magnitude. A pile oscillator experiment was performed in conjunction with Core 3 experiments,⁸ affording one determination of C_7 and λ_7 . Theoretically, three points on the experimental inhour curve also define ℓ_0 , C_7 , and λ_7 through Equation 4. This involves taking differences between numbers of the same order of magnitude, so that experimental errors became magnified to the point of rendering the calculations useless. A least squares fit using all of the data points available also did not yield results with acceptable confidence limits. It was therefore decided to employ calculated values of ℓ_0 and use the experimental inhour curve data to calculate C_7 and λ_7 . A detailed description of this process, as applied to Cores 1 and 2, along with the justification which exists for treating the reflector thermalized neutrons as a delay group, is given in Section V of Reference 2. The material presented there stands as the best information on the subject produced by the KEWB program. Within the bounds of experimental errors no difference in the values of the reflector delayed neutron parameters were evidenced in Core 3. Core 4 was not studied extensively because (for mechanical reasons) it could not be pulsed at periods shorter than about 2.5 msec, so that insufficient inhour curve data exist to define the reflector delayed neutron parameters. Since the change from the spherical to cylindrical core (11.45 and 13.65 to 18 ℓ loadings) did not produce a detectable change in these parameters, it was assumed that this would also be the case for Core 4.

Experiments with the reflector removed, Core 5, yielded experimental inhour data which are described by the theoretical equation with constant lifetime

and the conventional six groups of fission product delayed neutrons.* While the unreflected experiment does not provide proof of the validity of the seventh group treatment of the reflector effect, it does show that the effect is a property of the reflector, because omission of the reflector removed the observed anomalies in the inhour curve and reactor behavior.

Values of the reflector delayed neutron parameters, as well as those for all the parameters in the model, appear in Table VII on the next page. Figure 12 shows the experimental and theoretical inhour curves for Cores 3 and 5.

The graphite reflected reactor does not produce a burst with as short a period as the unreflected for a given reactivity release in excess of one dollar. As previously discussed for Cores 1 and 2 in References 1 and 2, this, in addition to being a consequence of the lifetime change imposed by the increase in uranium loading required for operation without the reflector, is the result of the storage of thermal neutrons in the reflector. That is, reflector thermalized neutrons require on the order of milliseconds to return to the core, and thus are temporarily lost to the reactor to the degree that the transient proceeds in the time scale of their mean delay time. This is important to safety considerations of the reactor type because it permits a larger maximum reactivity loading (to accommodate reduction in reactivity by irradiation samples or elevated temperature operation) with a fixed degree of safety in terms of the minimum possible reactor period in an accidental excursion.

2. Temperature Compensated Reactivity

The temperature coefficient of reactivity for the KEWB reactor core loadings was found to be a function of the fuel solution temperature, rather than constant, as reported earlier.² This is believed to be the result of nonlinearity in

*There was some deviation from this curve at reactor periods less than 1 msec. It was not possible to fit the deviation by addition of a seventh group of neutrons, as in the reflected reactor case. It is believed the effects observed were the result of reflection of neutrons from the transient inducing poison rod, which was a few inches away from the core as the stable period was developed in these very short period transients. During transient initiating procedures, the reactor is first made critical, then shut down by insertion of the poison rod. In the bare core, insertion of the poison rod caused a temporary positive period in the reactor, indicating the poison rod acted as a slight reflector when approaching the core. When fully inserted it of course shut the reactor down by a margin of several dollars.

TABLE VII
KINETICS MODEL PARAMETER VALUES*

Parameter	KEWB Reactor Core Loadings				
	Core 1	Core 2	Core 3	Core 4	Core 5
C	0.0308	0.0308	0.0308	0.0308	0.008
ℓ_o (μsec)	16	22	20.8	26	14.7
a ($\$/^\circ\text{C}^{-1}$)	-0.0166	-0.0327	-0.0269	-0.020	-0.0411
b ($\$/^\circ\text{C}^{-2}$)	-0.000247	-0.00022	-0.000457	-0.00058	-0.000519
D ($\$/\text{cm}^{-3}$)	-0.00476	-	-0.00206	-0.00358	-0.0354
C_7	0.0228	0.0228	0.0228	0.0228	0
λ_7 (sec^{-1})	500	500	500	500	0
k ($^\circ\text{C-Mj}^{-1}$)	20.0	16.9	13.6	9.40	10.2
E_o (Mj)	0.5	-	0.9	1.2	1.0
V_{core} (cm^{-3})	11,450	13,650	18,000	26,000	24,000
$A\delta h$ ($\text{cm}^4\text{-gm}^{-1}$)	19.1	-	27.1	19.4	18.5
g ($\text{psi-cm}^3\text{-Mw}^{-1}$)	8.3×10^3	-	8.3×10^3	8.3×10^3	8.3×10^3
τ_1 (sec)	0.0048	-	0.0048	0.0048	0.002

*For all core loadings, the following values of C_i and λ_i were used for fission product precursor delayed neutrons, from which the proper f_i are readily computed by Equation 13: $C_1 = 0.000264$, $C_2 = 0.001752$,

$$C_3 = 0.001568, C_4 = 0.00316, C_5 = 0.00092, C_6 = 0.000336, \sum_{i=1}^6 C_i = 0.008,$$

$$\lambda_1 = 0.01244, \lambda_2 = 0.0305, \lambda_3 = 0.1114, \lambda_4 = 0.301, \lambda_5 = 1.14, \lambda_6 = 3.01.$$

the coefficient of expansion of water. The density component of the temperature coefficient of reactivity should reflect this property directly. To determine the temperature coefficient of reactivity experimentally, data were obtained by non-nuclear heating of the reactor to a particular temperature, and, at thermal equilibrium, determining the control rod position for criticality. As this position varies with temperature, the control rod calibration curve provides the amount of reactivity that the system has lost. For the KEBW core loadings such

Figure 12. Experimental and Theoretical Inhour Curves, Cores 3 and 5

data were fitted very well by a quadratic,⁹ so that

$$R_T = A_0 + aT_c + \frac{b}{2} T_c^2 \quad \dots(15)$$

where R_T is the temperature compensated reactivity and T_c is the temperature in °C.

The temperature coefficient of reactivity is taken as the derivative of Equation 15

$$\frac{dR_T}{dT_c} = a + bT_c \quad \dots(16)$$

Equations 3 and 5 are written in terms of temperature increase during the transient, that is $T = \Delta T_c$. With this in mind the temperature compensated reactivity is

$$R_T = \int_{T_o}^{T_f} (a + bT_c) dT_c \quad \dots(17)$$

$$= a(T_f - T_o) + \frac{bT_f^2}{2} - \frac{bT_o^2}{2}$$

$$= a\Delta T_c + \frac{b}{2}(T_f - T_o)(T_f + T_o - T_o + T_o)$$

$$= a\Delta T_c + \frac{b}{2}\Delta T_c(\Delta T_c + 2T_o)$$

$$= (a + bT_o)T + \frac{b}{2}T^2. \quad \dots(18)$$

The temperature compensated reactivity as given by Equation 18 appears in Equation 3.

The experimental data for Core 3 are shown on the least square fit to the data in Figure 13.⁹ Similar curves for the other core loadings yield the values for a and b in Table VII, with the exception of those for Core 5. In that case the heating experiment temperature coefficient gave rise to much less temperature compensated reactivity than the total compensated reactivity in temperature controlled regions of bursts, as computed with the AIRCOMP code (Reference 2, Section I). The latter coefficient is deduced from Figure 14, where the total compensated reactivity for a number of bursts is plotted as a function of energy release. There the compensated reactivity is linear and essentially

Figure 13. Excess Reactivity as a Function of Fuel Solution Temperature, Core 3

Figure 14. Compensated Reactivity as a Function of Energy Release, Core 5

identical for all of the transients until an energy release in excess of 0.2 Mj is attained. Reference to Column 5 of Table VI and Figure 14 will show that the linearity is maintained until after peak power for all of these transients. Since the temperature rise is small the quadratic term of the temperature coefficient of reactivity is negligible for the values of energy release under discussion. The curves for transients 3047, 3049, 3050, 3051, and 3052 differ from the other curves when the energy is larger for reasons which will be discussed later. The linear slope of the curves gives

$$a + bT_o = \frac{\Delta R}{k \Delta E} = \frac{\Delta R}{\Delta T} = 0.0534 \text{ } \$/^{\circ}\text{C.} \quad \dots(19)$$

This differs markedly from the heating experiment value, $0.0301 \text{ }^\circ\text{C}^{-1}$ for T_0 equal to 25°C . Two possible explanations of the discrepancy are:

- a) The reciprocal heat capacity or the power calibration and hence the energy release for this core loading were in error.
- b) The heating experiment was, through error or physical process, inapplicable to the dynamic temperature coefficient.

With regard to the first possibility, the reciprocal heat capacities used for all the core loadings have been calculated assuming the solution volume is all water. These dilute aqueous solutions could not have an appreciably larger reciprocal heat capacity than water.

More convincing evidence of the consistency of the reciprocal heat capacity and the power calibration is available from the data in Table VI. The average of the product of the total energy release, which reflects the power calibration in that the energy is the integral of the power curve, in Column 4, and the reciprocal heat capacity, $10.2^\circ\text{C}/\text{Mj}$, will be found to be equal to the average temperature rise, Column 10 minus Column 6 within 1%. We therefore conclude that the power calibration and reciprocal heat capacity were not in error for this core loading.

Although no concrete justification is available, the second hypothesis covers more than one possibility which could enable one to discount the applicability of the heating experiment results. One is that an error might have been made in the experiment or recording of data, although the checks that could be made did not indicate this is the case. Another is that the heating experiment method contains errors which were of second order when the reflector was present but were important in the unreflected reactor. This would be consistent with the neutronic sensitivity of the unreflected reactor, which definitely exceeded that of the other core loadings.

Since we are unable to show conclusive evidence of the inapplicability of the coefficient indicated by the heating experiment to the dynamic reactor but are convinced that we have a faithful report of the coefficient actually at work in the reactor, we are left with no choice but to adopt the latter.

A value for the quadratic element of the coefficient was obtained from an estimate of the average position of the curves for transients 3047, 3049, and 3052 at higher values of energy in Figure 14. These, and transients 3050 and

3057 were run with an initial core pressure of 11 atm to eliminate void formation after peak power in these transients. The shutdown mechanism for these transients thus consists only of temperature compensation of reactivity. The sudden curvature and rapid increase in the compensated reactivity curves for the other transients shown in Figure 14 is the result of radiolytic gas void formation. Elimination of this characteristic with high pressure is an important corroboration of the validity of the temperature and void shutdown mechanisms ascribed to the reactor.

3. Radiolytic Gas Void Growth

The radiolytic gas void model described by Equations 6 to 10 is explained in detail and parameter values are given in Reference 3. A brief reiteration of the physical picture of the void growth is as follows: Until the threshold energy release is reached, indicative of a certain high degree of dissolved hydrogen gas supersaturation in the fuel solution, bubbles nucleated by fission centers and fragments quickly collapse. At E_0 , however, these and new bubbles are able to accumulate gas in an extremely short time. This gives rise to an inertial pressure, providing the driving force for the growth of void volume. Equation 8 provides the value of E_0 ; Equations 6 and 7 provide means for computing the void volume from the pressure, and Equations 9 and 10 define the pressure pulse.

The model of the pressure pulse is the least rigorous of the components given. This is so because the detailed microscopic picture of the void process is complicated by the time spread in reaching the threshold energy across the core (because of the flux peak) and by lack of information on the interaction of the bubbles in competing for dissolved gas. The pressure pulse equations consist essentially of means for predicting the peak pressure, deduced from theoretical considerations and verified by the reactor data, and semi-empirical exponential functions which shape the pulse. The peak pressure in the reactor was found to be proportional to the power density in the core with a unique constant of proportionality within a few percent for all of the KEWB core loadings except Core 2.* The rise and decay exponentials are justified in Reference 3.

*In Core 2 the constriction of the core neck limited the free surface area of the solution, inhibiting expansion, and causing higher pressures than corresponding transients in the other cores. Because of this complication the void model is not applicable to Core 2.

In particular, note Appendix III in this reference for another model which enables calculations of the pressure exponential relaxation period. One feature included in Equations 9 and 10, although perhaps not improving the model appreciably, is the division of the pressure pulse into two parts, one rising and the other decaying, with the pressure peak made to occur 1.5 periods after the onset of the pressure pulse. This was incorporated because the experimental pressure rises to its peak in ~ 1.5 periods for short period transients (2 or 3 msec and less).

The void formation threshold energy release for the unreflected reactor was 1.0 Mj. The compensated reactivity for the unpressurized transients in Figure 14, however, shows the rapid change in compensated reactivity characteristic of radiolytic gas void formation at much lower values of energy release. In fact, these bursts never attain an energy release of 1.0 Mj. The periods of these transients were in tens and hundreds of msec rather than msec, where the inertial pressure makes its appearance. The void model thus does not apply to transients with periods in excess of about 10 msec. The reason for this probably lies in the longer time scale of these transients, which may permit void growth in a less highly supersaturated solution. The given model has been found to be moderately descriptive of the reactor at periods from 10 to 5 msec, better at periods from 5 to 3 msec, and a good description from 3 to the shortest periods run, about 0.6 msec. Since the last group is the one of prime concern, namely, that in which the inertial pressure peak is in hundreds of psi, the model serves its purpose in being most accurate in this period region.

4. Tabulation of Kinetics Model Parameter Values

Values of all of the parameters which must be specified for the kinetics model, excepting initial conditions, Category 1 at the beginning of this section (III-C) are given in Table VII. The absence of void model parameters for Core 2 is a result of the inapplicability of the model to this loading for the reasons footnoted in the section preceding this.

D. ANALYTIC RESULTS AND COMPARISONS WITH EXPERIMENTS

1. Analysis of Data with the AIRCOMP Code

The kinetic response of reactor is defined by the time course of the effects of the reactivity compensating and neutron storage (delayed neutron) mechanisms on the input reactivity. While such data were produced by the AIRCOMP program² for all of the transients with energy release entries in

Tables II to VI, a digest of this information cannot be presented concisely. Considerable insight into the kinetic behavior of a reactor can be gained, however, from a study of the reactivity distribution at the time of peak power. This data is presented for Core 3 in Figure 15. There was not as much data for the other core loadings, and those available essentially repeat, in a qualitative sense, the information for Core 3. At peak power $dN(t)/dt = 0$, so that, from Equation 1

$$R(t) - 1 = \frac{1}{N(t)} \sum_{i=1}^n f_i W_i(t) \quad \dots (20)$$

where the index n is over six or seven groups of delayed neutrons, as appropriate. If the reactor period, τ_o , is such that $\frac{1}{\tau_o} \gg \lambda_i$, the reactivity at peak power is one dollar. This is the case for the fission product delayed neutrons at short periods in the KEWB reactor, hence the asymptotic approach to one dollar of the reactivity stored in fission product delayed neutrons, R_{1-6} , in Figure 15. The reactivity stored in reflector delayed neutrons, R_7 , if present, is the seventh term in the summation in Equation 20. R_c is the reactivity in all the terms on the right hand side of Equation 3, except R_o , R_T is the sum of the temperature compensated reactivity terms in Equation 3 (stated explicitly in Equation 18), and, since the formulation for R_V is not assumed to be known in the AIRCOMP analysis, R_V is the difference between R_c and R_T .

We see from Figure 15 that the void shutdown mechanism is not as important as temperature in turning over the burst. This is because the threshold energy release is not reached, and thus void is not formed until near peak power in any of the short period bursts. The void growth is so precipitous, however, that it sharply decreases the power after the peak, as will be seen in bursts shown later, especially those for the unreflected reactor. We also see from Figure 15 that, at least until peak power, the reactor is essentially completely shut down by temperature compensation of reactivity when the reactor period is greater than perhaps 20 msec.

Figure 15. Input Reactivity and Reactivity Distribution at Peak Power as a Function of Reactor Period, Core 3

Not shown in Figure 15 is the reappearance of void compensated reactivity for reactor periods greater than several hundreds of milliseconds. This void is believed to be the result of ordinary diffusion of radiolytic gas from a slightly or mildly supersaturated solution into bubbles formed on nucleation sites such as dust and cooling coil surfaces. It is present until the bubble residence time, which is believed to be of the order of seconds, becomes short relative to the reactor period. At still longer periods the reactor becomes temperature controlled again. Further discussion of these phenomena may be found in References 1 and 2. Since transients at periods of hundreds of milliseconds and longer are relatively mild and present no hazardous safety problems, the formation of void at long periods has been considered of academic interest only in the KEWB program and therefore has not been subjected to detailed analysis.

2. The AIREK Code

Solution of the kinetics model defined by Equations 1 through 10 is provided by a special version of the AIREK computer program, described in complete detail in Reference 4. This program has been used to produce the theoretical bursts to be presented here.

The AIREK program contains provision for a neutron source term as an initial condition, a factor which has been omitted from the model in this report, for convenience. This program is used for theoretical step reactivity input bursts by definition of an input reciprocal period (or a value of the reactivity, if desired), initial temperature, power level, and the appropriate values for the parameters specified in Table VII. The program assumes the reactor has been on the specified period from an infinitesimally small power level, computes values for the relative delayed neutron concentrations, and then computes the burst as dictated by the equations and parameter values.

In that the void model is probably inapplicable to ramp reactivity input induced bursts, so also would be the program. Ramp induced excursions however, are relatively mild, because of the limited ramp rates possible with control rod drive mechanisms. The most severe produced in the KEWB program

are comparable to temperature controlled step bursts with periods in the tens of msec range. The program has provisions for ramp reactivity input and if the void parameters are omitted, ramp excursions with temperature shutdown only may be studied.

3. Comparison of Experimental and Theoretical Temperature Controlled Excursions

The predominantly temperature controlled region of KEWB reactor bursts permits us to study the kinetics model with the void parameters omitted. The reactor power in two typical theoretical bursts computed with the AIREK code for Core 3 parameters are shown in Figures 16 and 17 along with the experimental bursts at the same period. These are seen to be in excellent agreement, until after the power peak.

Figure 16. Experimental and Theoretical Power for a Reactor
Period of 23 msec, Core 3

At the end of the KEWB experimental program the tests mentioned in Section III-C-2 were performed with the unreflected core under 11 atm of

Figure 17. Experimental and Theoretical Power for a Reactor
Period of 83 msec, Core 3

pressure to suppress the void and obtain bursts controlled by temperature only. The appropriate theoretical bursts with no void feedback were computed with the AIREK program. Two of these are shown with the experimental power data in Figures 18 and 19. These indicate even more conclusively than the material in Figures 16 and 17 the validity of the temperature shutdown portion of the kinetics model.

4. Comparison of Experimental and Theoretical Short Period Excursions

For the short period transients in which radiolytic gas void is present many theoretical bursts with the appropriate parameter values from Table VII were computed with the AIREK program for Cores 1, 3, 4, and 5. Figures 20, 21, and 22 show the theoretical and experimental power, void compensated reactivity and inertial pressure respectively for a Core 1 burst. Since experimental data and AIREK calculations have different arbitrary time origins, the curves of power shown have been shifted to make the bursts originate simultaneously. No

Figure 18. Experimental and Theoretical Power for
a Reactor Period of 260 msec, Core 5
Pressurized

Figure 19. Experimental and Theoretical Power for a Reactor
Period of 27 msec, Core 5 Pressurized

Figure 20. Experimental and Theoretical Power for a Reactor Period of 2.3 msec, Core 1

further alteration was made for the void compensated reactivity and pressure curves; that is, the good agreement in time is provided by the model. The same is true for Figures 23 to 37. The experimental power, and hence void compensated reactivity for Core 1 bursts were distorted by the high degree of gamma radiation sensitivity present in the neutron detectors,⁵ and therefore the comparison after peak power is not meaningful. Also, the experimental pressure in

Figure 22 shows the perturbations in filling the sphere after the peak pressure, which was not included in the theoretical model.

Figures 23 to 31 show the theoretical and experimental peak power, void compensated reactivity and inertial pressure successively for each of three Core 3 short period transients, with results generally in excellent agreement. Similarly, Figures 32 to 34 show the data for one Core-4 transient, and Figures 35 to 37 show the data for one Core-5 transient. To further illustrate the excellent agreement between the theoretical results and experiments, values of the computed peak power, energy release to peak power, total energy release and peak inertial pressure are plotted on duplicates of portions of Figures 6, 10, 9, and 7 in Figures 38 to 41 respectively, for each of three Core 1, six Core 3, four Core 4, and six Core-5 bursts.

5. Ramp Induced Transients

Reactor excursions induced by the ramp input of reactivity are important to hazards analysis because of the greater likelihood of their occurrence than step induced transients. For the maximum possible ramp rates up to \$0.16/sec and maximum reactivity insertions to at least six or seven dollars, excursions produced in Cores 1 and 2 were almost totally shut down by temperature compensation of reactivity. At worst, they were not more severe than step induced transients with periods of about 50 msec. The AIREK code contains provision for ramp reactivity input, and it may be employed to calculate such excursions with the void model parameters omitted. Since no inertial pressure was observed in ramp transients, in a continuous rod withdrawal accident the reactor will suffer no adverse consequences. Consideration of the possibility of greater ramp rates is of academic interest: higher withdrawal speed control rods would be required but are not necessary for operation of the reactor, and would be difficult to design. In the event of the desire to perform such analysis, however, it is certain the ramp transient will not be more severe than the equivalent step transient induced by the total reactivity release. This defines the conservative limit. To approximate how much less severe it would be the ramp case may be computed, but with the restriction of the uncertainty of applicability of the void model in the event that the burst positive instantaneous period is other than about 5 msec or less when the threshold energy for void formation is reached.

CORE 1, REACTOR PERIOD = 2.25 MS

7723-10
054 000

Figure 21. Experimental and Theoretical Void Compensated Reactivity for a Reactor Period of 2.3 msec, Core 1

Figure 22. Experimental and Theoretical Pressure for a Reactor Period of 2.3 msec, Core 1

CORE 1, REACTOR PERIOD = 2.25 MS

7723-10
055 000

Figure 23. Experimental and Theoretical Power for a Reactor Period of 2.9 msec, Core 3

CORE 3, REACTOR PERIOD = 2.90 MS

7723-10
012 000

Figure 24. Experimental and Theoretical Void Compensated Reactivity
for a Reactor Period of 2.9 msec, Core 3

CORE 3, REACTOR PERIOD = 2.90 MS

7723-10
013 000

Figure 25. Experimental and Theoretical Pressure for a Reactor Period of 2.9 msec, Core 3

Figure 26. Experimental and Theoretical Power for a Reactor Period of 1.4 msec, Core 3

CORE 3, REACTOR PERIOD = 1.40 MS

7723-10
019 000

Figure 27. Experimental and Theoretical Void Compensated Reactivity for a Reactor Period of 1.4 msec, Core 3

CORE 3 REACTOR PERIOD = 1.40 MS

7723-10
020 000

Figure 28. Experimental and Theoretical Pressure for a Reactor Period of 1.4 msec, Core 3

Figure 29. Experimental and Theoretical Power for a Reactor Period of 1.0 msec, Core 3

CORE 3, REACTOR PERIOD = 1.00 MS

7723-19
026 000

Figure 30. Experimental and Theoretical Void Compensated Reactivity
for a Reactor Period of 1.0 msec, Core 3

CORE 3, REACTOR PERIOD = 1.00 MS

7723-10
027 000

Figure 31. Experimental and Theoretical Pressure for a Reactor Period of 1.0 msec, Core 3

Figure 32. Experimental and Theoretical Power for a Reactor Period of 2.5 msec, Core 4

Figure 33. Experimental and Theoretical Void Compensated Reactivity
for a Reactor Period of 2.5 msec, Core 4

CORE 4, REACTOR PERIOD = 2.50 MS

7723-19
076 90c

Figure 34. Experimental and Theoretical Pressure for a Reactor Period of 2.5 msec, Core 4

Figure 35. Experimental and Theoretical Power for a Reactor Period of 1.5 msec, Core 5

CORE 5, REACTOR PERIOD = 1.50 MS

7723-10
103 990

Figure 36. Experimental and Theoretical Void Compensated Reactivity
for a Reactor Period of 1.5 msec, Core 5

CORE 5, REACTOR PERIOD = 1.50 MS

7723-10
104 04C

Figure 37. Experimental and Theoretical Pressure for a Reactor Period of 1.5 msec, Core 5

Figure 38. Experimental and Theoretical Peak Power as a Function of Reactor Period, Cores 1, 3, 4, and 5

Figure 39. Experimental and Theoretical Energy Release to Peak Power as a Function of Reactor Period, Cores 1, 3, 4, and 5

Figure 40. Experimental and Theoretical Total Energy Release as a Function of Reactor Period, Cores 1, 3, 4, and 5

Figure 41. Experimental and Theoretical Peak Inertial Pressure as a Function of Reactor Period, Cores 1, 3, 4, and 5

E. USE OF THE KINETICS MODEL FOR OTHER AQUEOUS HOMOGENEOUS REACTORS

Standard nuclear calculations will furnish values for the reactor parameters in Categories 2 and 3 on page 33. The reflector delayed neutron parameters for the KEWB reactor may be employed for 12-in. diam water boilers with nuclearly infinite graphite reflectors. Good results have been obtained in the calculation of these parameters with multigroup multiregion diffusion codes, the lifetimes being defined with appropriate poisoning calculations. Other interesting material on the subject exists in the literature.¹¹

For the void model, the parameter $g = 8.3 \times 10^3$ psi-cm³/Mw, as in Table VII. The quantity τ_1 can be taken to be 0.0048 with little chance for appreciable error or can be computed (with some difficulty) using the model given in Appendix III in Reference 3. The other parameter values are simple system constants or quantities for which computational instructions are included in Reference 3. The AIREK computer program in Fortran language is completely described in Reference 4, with program decks available from Atomics International. Thus, all of the information necessary to use the kinetics model should be available.

Since the void model data apply to a wide range of aqueous homogeneous reactor fuel concentrations and solution volumes in the KEWB core loadings, and no restrictions for step input transients other than limitation of fuel solution free surface area are placed on the applicability of the model in Reference 3, it is safe to assume that the model will predict with accuracy the kinetic behavior of aqueous homogeneous reactive assemblies differing markedly in configuration from the KEWB reactor. This will be the case in particular in the period region of high inertial pressure, which is of the greatest concern for reactor safety evaluations. Some error in prediction will be present after peak power for bursts with reactor periods greater than 10 msec, but good accuracy will hold at least to the time of peak power for excursions with periods up to several hundreds of msec.

F. CONCLUSIONS

A step reactivity input kinetics model has been presented and shown to compare very favorably with experimental data from four KEWB reactor core loadings. This is particularly remarkable in that many possible theoretical complications have deliberately been avoided in construction of the model. It can provide increased reliability in safety studies and hazards evaluations of existing and future aqueous homogeneous research reactors. It also should prove a valuable tool in the study of chemical reprocessing plant accidents. Its application there could eliminate the need for a multitude of expensive nuclear safety experiments and provide cost savings by avoiding unnecessary overdesign of plant and vessels.

REFERENCES

1. J. W. Flora and R. K. Stitt (ed.), "Kinetic Experiments on Water Boilers 'A' Core Report — Part I, Program History Facility Description and Experimental Results," NAA-SR-5415 (March 1962)
2. M. Dunenfeld (ed.), "Kinetic Experiments on Water Boilers 'A' Core Report — Part II, Analysis of Results," NAA-SR-5416 (February 1962)
3. P. Spiegler, C. F. Bumpus, Jr., and A. Norman, "Production of Void and Pressure by Fission Track Nucleation of Radiolytic Gas Bubbles During Power Bursts in a Solution Reactor," NAA-SR-7086 (in press)
4. M. Hoffman and L. R. Blue, "Program for the Numerical Solution of the Space Independent Reactor Kinetics Equations for KEWB," Applied Math. Technical Document 132, Atomics International (November 1962)
5. E. L. Gardner, "The Development of Linear Power Instrumentation for the Kinetic Experiments on Water Boilers," NAA-SR-7077 (in press)
6. E. L. Gardner, "Instrumentation for Experimental Reactor Studies," Communication and Electronics (AIEE) 50 (September 1960)
7. M. E. Remley, J. W. Flora, D. L. Hetrick, and L. P. Inglis, "Program Review of the Water Boiler Reactor Kinetic Experiments," NAA-SR-1525 (March 15, 1956)
8. R. N. Cordy, "Kinetic Experiments on Water Boilers — 'A' Core Report — Part III — Pile Oscillator Results," NAA-SR-5417 (December 1960)
9. P. Spiegler (in an Atomics International internal document)
10. M. Greenfield, "Kinetic Experiments on Water Boilers 'A' Core Report — Part IV — Containment Aspects of Hydrogen-Oxygen Explosions," NAA-SR-5418 (December 1960)
11. C. E. Cohn, "Reflected-Reactor Kinetics," Nuclear Science and Engineering 13, No. 1 (May 1962)