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ORNL NUCLEAR SAFETY RESEARCH AND DEVELOPMENT PROGRAM BIMONTHLY REPORT FOR JULY-AUGUST 1967

Wm. B. Cottrell

ABSTRACT

The accomplishments during the months of July and August in the research and development program under way at ORNL as part of the U.S. Atomic Energy Commission's Nuclear Safety Program are summarized. Included in this report are work on various chemical reactions, as well as the release, characterization, and transport of fission products in containment systems under various accident conditions and on problems associated with the removal of these fission products from gas streams. Although most of this work is in general support of water-cooled power reactor technology, including LOFT and CSE programs, the work reflects the current safety questions, such as measurements of the prompt release phenomena and the efficacy of containment spray systems. Several projects are also conducted in support of the high-temperature gas-cooled reactor (HTGR) program, including both in-pile and out-pile studies of the reaction rates and fission-product release and transport phenomena. Other major projects include fuel transport safety investigations, a series of discussion papers on various aspects of water reactor technology, anti-seismic design of nuclear facilities, and studies of piping and pressure vessel technology. Experimental work relative to pressure vessel technology includes investigations of the attachment of nozzles to shells and the implementation of a joint AEC-PVRC program on heavy-section steel technology and nuclear piping, pumps, and valves. Several of the projects are directly related to another major undertaking, namely the AEC's standards program, which entails development of engineering safeguards and the establishment of codes and standards for government-owned or -sponsored reactor facilities. A project has recently been initiated to investigate safety problems of the molten-salt breeder reactor. Another task (CHORDS) is concerned with the establishment of computer programs for the evaluation of reactor design data. The recent activities of the NSIC and the <u>Nuclear Safety</u> journal in behalf of the nuclear community are also discussed.

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CONTENTS

1. WATER REACTOR PROGRAM — CHEMICAL REACTIONS

1.1 CHARACTERIZATION OF SIMULATED ACCIDENT AEROSOLS FOR CSE (AEC Activity 04 60 10 01 1)

G. W. Parker T. H. Row

The validity of simulated fission-product aerosols as compared with aerosols from highly irradiated UO_2 has been a question of much concern to the Nuclear Safety Research Program. One method of simulating such an aerosol is that provided for the Containment Systems Experiment at Battelle-Northwest in which small high-temperature furnaces are used to vaporize the desired amount of a fission-product element in a suitable flowing gas stream. The vaporized material then passes through a very hot furnace containing melted UO_2 before the mixed gases are conducted into the containment system.

The incentive for use of such simulants is that an impractically large radioactive source would be required if a suitable amount of reactor fuel were to be heated to melting. For the sake of comparing behavior of both the simulant and highly irradiated UO_2 in the same facility, we have conducted a small program on simulant behavior that includes a miniature version of the BNWL method of aerosol generation. Extremely small, densely compacted pellets containing suitable radioactivities and the required amount of nonradioactive fission-product elements are heated on small ribbons or filaments by low-voltage resistance. The vapors are then passed through a hot zone containing melted UO_2.

We have completed two runs that simulated fuel with burnup levels of 1000 and 7000 Mwd/T. These are designated in Table 1.1 as CMF runs 1007 and 1008. Technical difficulties have limited our success in generating suitable fractions of elements other than iodine, cesium, and ruthenium; however, a fair amount of data has been accumulated with these three elements. We have compared the results of the two runs in Fig. 1.1 and in Table 1.1. The major differences in amount of activity remaining airborne are attributed to the effect of having a delay in run 1008 that allowed most of the steam to condense before the aerosol was generated.

A secondary bonus from run 1007 was that two methods of sampling were used in parallel. It was interesting to observe the complete absence of the short half-lived component in the deposition of radioiodine, as seen

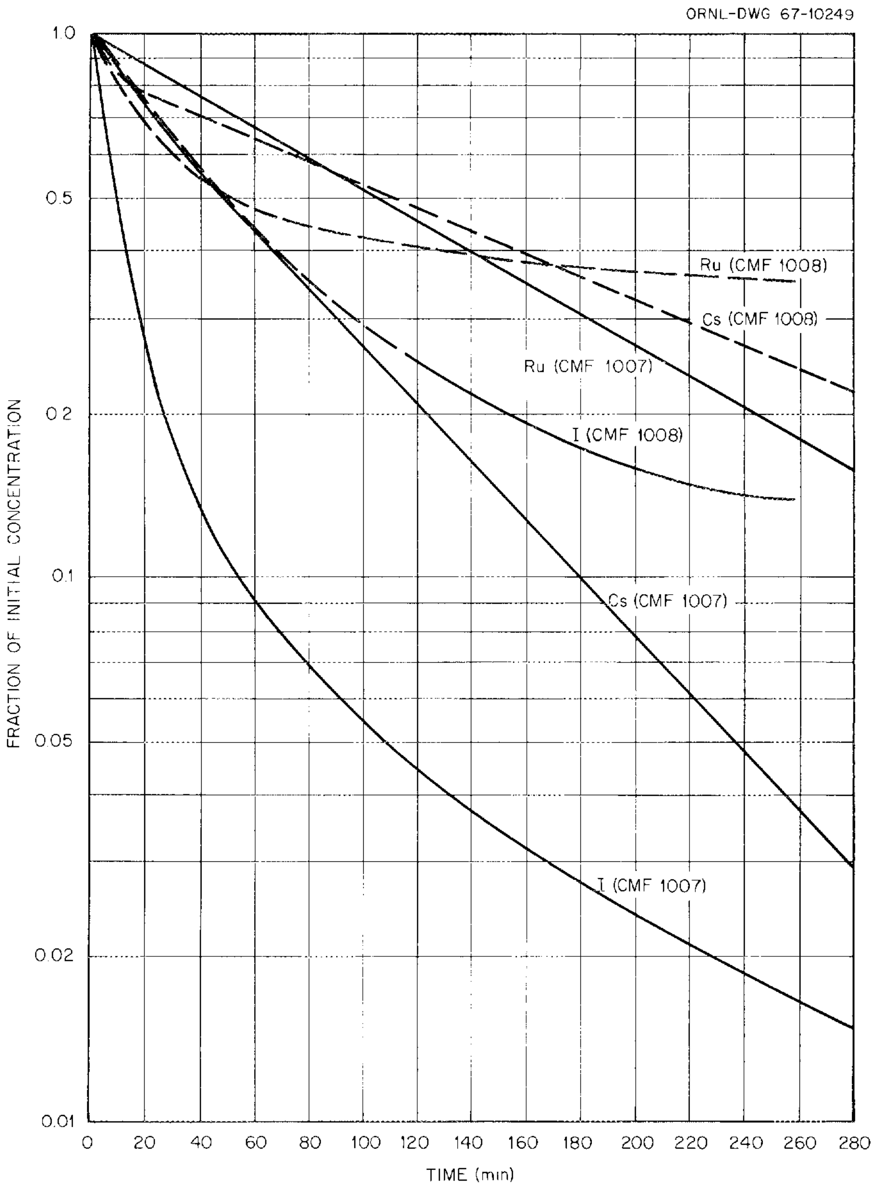

Fig. 1.1. Airborne Activity in Containment Tank as Indicated by Gas Samples.

Table 1.1. Comparison of Activity Half-Times Obtained in Two Simulant Runs

	$T_{1/2}$, Half-Time (min)	
	CMF Run 1007	CMF Run 1008
For steam condensation[a]		
From I_2 activity in condensates	121	174
From pressure-temperature readings	126	64.2
From condensate volumes	118	187
For cesium activity in gas samples	59	
For ruthenium activity in gas samples	92	152
For iodine activity in gas samples	18 and 128	28 and 264
For gross activity from ion chambers		
On top of CMF	20.6 and 619	67.2
On side of CMF	22.5 and 730	55

[a]Total volume of condensate was 216 ml in CMF 1007 and 115 ml in CMF 1008.

in Fig. 1.1, CMF run 1007, in data obtained by use of a sampler that was connected with only a short (18-in.) length of piping to the containment vessel. The other sampler from which the data in Fig. 1.1 were taken was actually inside the vessel; thus entrance effects were minimized.

The planned program for validation of the CSE simulant includes relocation of the equipment and melting highly irradiated UO_2 in the CRI facility. Comparative sampling will also be conducted in this case.

Three Zircaloy-clad UO_2 capsules have been inserted into the ORR reactor for long-term irradiations. The capsules are scheduled for a burnup of greater than 10,000 Mwd/T for use in subsequent CSE verification tests. Work is progressing on the design of an in-pile CSE simulant generator.

A portable sampling device was designed to be used in experiments conducted at CSE and CMF to aid in comparing the date from each facility. The portable sampling device was used to sample ADF run S-85 at BNWL Containment Systems Experiment facility. The run was isothermal at 80°C and 1 atm abs. The atmosphere was about a 50% steam–50% air mixture by volume. The aerosol consisted of iodine with ^{131}I tracer, cesium with ^{137}Cs tracer, and ruthenium with ^{103}Ru tracer. These isotopes were volatilized and passed over a fused zirconium-clad UO_2 fuel element before entering the SAT tank.

The samples from the experiment were analyzed with CSE counting equipment and the gamma-spectrum analysis program. The results have been received and will be interpreted during the next report period.

1.2 PROMPT FISSION-PRODUCT RELEASE STUDIES IN TREAT (AEC Activity 04 60 10 01 1)

G. W. Parker R. A. Lorenz

Work is continuing on the design of experiments to simulate in the TREAT reactor the "prompt" fission-product release phase of loss-of-coolant accidents. "Prompt release" refers to the immediate escape of volatile and gaseous fission products from void spaces in high-burnup fuel rods when cladding is ruptured following a loss-of-coolant blowdown accident in pressurized-water and boiling-water reactors. Cladding rupture in these accidents is a result of high internal gas pressure and low cladding strength because of high temperature following loss of coolant. In the in-pile TREAT experiments, fission heat will be used to simulate the decay heat and initial heat content of UO_2 above the cladding temperature in a real loss-of-coolant accident. TREAT will be operated at steady power so that cladding rupture of the test rods will occur between 10 and 30 sec after initial reactor heating. In the first experiments, it is planned to use clusters of VBWR fuel rods (Zircaloy-2 clad UO_2) irradiated to 9000 Mwd/T several years ago.

The prompt release phase of the loss-of-coolant accident poses two important questions to the reactor safety analyst. First, how much radioactive and biologically hazardous fission-product material will be released (and how far will it be transported). Second, will the cladding swell or fail in such a way that emergency core cooling devices cannot function properly.

The TREAT experiments will be concerned primarily with evaluation of the quantity and character of released radioactive fission products, especially ^{131}I. To provide background information for design of the experiments and analysis of results, a literature survey has been undertaken. The main objective is to look for correlation between fuel operating conditions, the "diffusion" release of short half-life isotopes, and the release of stable fission gases. Much work has been done on the release of stable fission gas under normal fuel operating conditions, but there is little agreement on correlation or prediction. The only study of short half-life isotopes was made with continuous sweep experiments, usually at low burnup and low temperature.

In our attempt to correlate the release of radioactive and stable fission gas with UO_2 fuel operating conditions, we are examining the use of the "classical" diffusion release equations developed by Booth and Rymer[1,2] and also by Beck.[3] These equations describe the release of both radioactive and stable fission gas. The equations are

$$F = 4\sqrt{D' t/\pi} \qquad \text{and} \qquad F = 3\sqrt{D'/\lambda}$$

for small fractional release of stable and radioactive fission products, respectively. Figure 1.2 is a graph of the two pertinent equations. It is now known that fission-gas release, especially at high temperature, is dependent on a variety of parameters, such as burnup, grain growth, trapping, bubble migration, and possibly temperature cycling. However, we believe that the time dependence and radioactive decay dependence described in the diffusion equations may be valid, and we are therefore examining the use of a fission-gas release parameter, D', whose temperature dependence is empirically determined. In agreement with Evans[4] we will call the parameter D' (empirical), and it is to be substituted for D/a^2 or D' in the "classical" diffusion equations.

The temperature dependence of fission-gas release in capsules and fuel rods has been described in a variety of ways. Examples of temperature-equivalent parameters used to correlate fractional fission-product

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[1]A. H. Booth and G. T. Rymer, Determination of the Diffusion Constant of Fission Xenon in UO_2 Crystals and Sintered Compacts, Canadian Report CRDC-720, August 1958.

[2]J. Belle (Ed.), Uranium Dioxide: Properties and Nuclear Applications, p. 486, U.S. Government Printing Office, Washington, 1961.

[3]S. D. Beck, The Diffusion of Radioactive Fission Products from Porous Fuel Elements, USAEC Report BMI-1433, Battelle Memorial Institute, Apr. 18, 1960.

[4]D. M. Evans, The Use of a Parameter to Characterise the Release of Stable Fission Gases in Sealed Pins Operating in Power Reactors, British Report TRG-969(W), 1965.

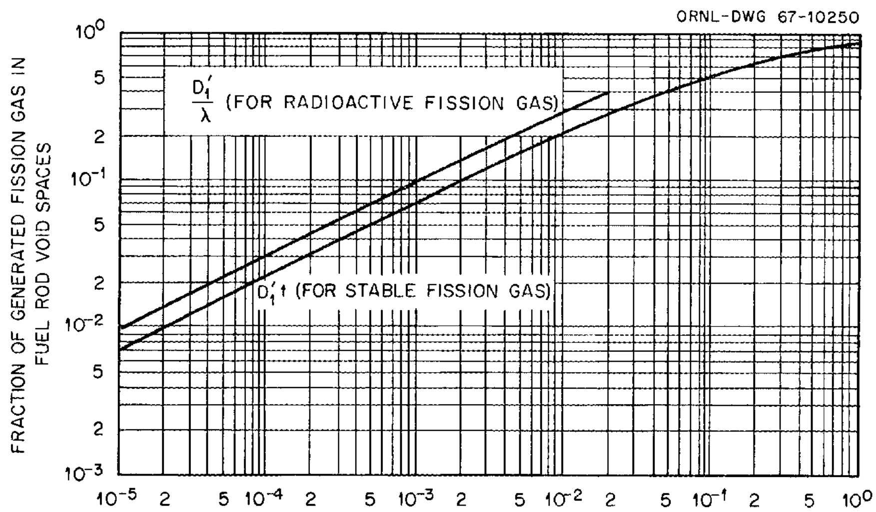

Fig. 1.2. Graph of Expressions Describing Release of Radioactive and Stable Fission Gas from UO_2.

release are volumetric average fuel rod temperature,[5] maximum fuel rod volumetric average temperature,[6] the thermal conductivity integral $\int_s^c K\, d\theta$ at end of experiment,[7] and the integral $\int_{400}^{c} K\, d\theta$ averaged over time and length of rod.[8]

The choice for our first attempt at determining D′ (empirical) was to use the release of stable fission gas in capsules and fuel rods as reported by AECL.[8-12] We chose the AECL work because of the large number of carefully conducted and well-reported experiments. The temperature-equivalence we chose is the average linear power, which was easily obtained from most of the AECL reports. This average linear power can be obtained for other irradiations by using burnup data and time at power and by assuming heat output of 182 Mev per fission. A small correction was made for density differences based on Ref. 12. The resulting D′ (empirical), which will be referred to as D_1', is correlated with average

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[5]M. D. Freshley and F. W. Panisko, The Irradiation Behavior of UO_2-PuO_2 Fuels in PRTR, USAEC Report BNWL-366, Battelle Northwest Laboratories, March 1967.

[6]J. P. Hoffman and D. H. Coplin, The Release of Fission Gases from Uranium Dioxide Pellet Fuel Operated at High Temperatures, USAEC Report GEAP-4596, General Electric Company, September 1964.

[7]R. C. Nelson, D. H. Coplin, M. F. Lyons, and B. Weidenbaum, Fission Gas Release from UO_2 Fuel Rods with Gross Central Melting. I. Pellet Fuel, USAEC Report GEAP-4572, General Electric Company, July 24, 1964.

[8]R. D. MacDonald and G. W. Parry, Zircaloy-2 Clad UO_2 Fuel Elements Irradiated to a Burnup of 10,000 Mwd/Tonne U, Canadian Report AECL-1952, January 1964.

[9]M. J. F. Notley, A. S. Bain, S. Ananthakrishnan (Atomic Energy Establishment, Trombay, India), and G. W. Parry, Zircaloy Sheathed UO_2 Fuel Elements Irradiated at Values of $\int K d\theta$ Between 40 and 83 W/cm, Canadian Report AECL-1676, December 1962.

[10]A. S. Bain, J. Christie, and A. R. Daniel, Trefoil Bundles of NPD 7-Element Size Fuel Irradiated to 9100 Mwd/tonne U, Canadian Report CRFD-1179 or AECL-1895, January 1964.

[11]R. D. MacDonald and A. S. Bain, Irradiation of Zircaloy-2 Clad UO_2 to Study Sheath Deformation, Canadian Report AECL-1685, December 1962.

[12]M. J. F. Notley and J. R. MacEwan, The Effect of UO_2 Density on Fission Product Gas Release and Sheath Expansion, Canadian Report AECL-2230, March 1965.

linear power in Fig. 1.3. D_1' is the empirical fission-gas release parameter for stoichiometric UO_2, 93.5% of theoretical density, with burnup greater than 380 Mwd/T, derived from a correlation of released stable fission gas with average linear power based on burnup, time at power, and 182 Mev per fission. The capsules were subjected to temperature cycles, but the operating power was reasonably constant for the duration of the experiment.

The method for determining D_1' for each point was to find $D_1't$ from Fig. 1.2, divide by irradiation time at power, and apply a small correction for UO_2 density. Actually the density correction should more logically be applied to the linear power.

The results are quite gratifying because good agreement is obtained over a large range of time, burnup, and density. Note that the present linear power correlation is good only for "normal" fuel-to-cladding conductivity. An approximate volumetric-average temperature scale is included at the top of Fig. 1.3 for rough comparison.

The next step is to apply D_1' to the radioactive fission gases that are in production-decay equilibrium. As an example, a section of fuel rod operating at 18.3 kw/ft (600 w/cm) would have a D_1' of about 2×10^{-9} sec^{-1} (from Fig. 1.3). The fraction of ^{133}Xe ($\lambda = 1.52 \times 10^{-6}$ sec^{-1}) in the fuel rod void spaces would be 0.11 or about 11% (from Fig. 1.2). If iodine fractional release is twice that of xenon,[13] then D_1' for iodine would be $4(2 \times 10^{-9})$, or 8×10^{-9} sec^{-1}, and the fraction of ^{131}I ($\lambda = 10^{-6}$ sec^{-1}) in the fuel rod voids at production-decay equilibrium would be about 27% (Fig. 1.2) for the above example. It is not possible to check this estimate because very few experimental values are available on the release of short half-life fission products from high-burnup fuel.

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[13] G. W. Parker and R. A. Lorenz, ORNL Nuclear Safety Research and Development Program Bimonthly Report for May—June 1967, USAEC Report ORNL-TM-1913, p. 6, Oak Ridge National Laboratory, July 1967.

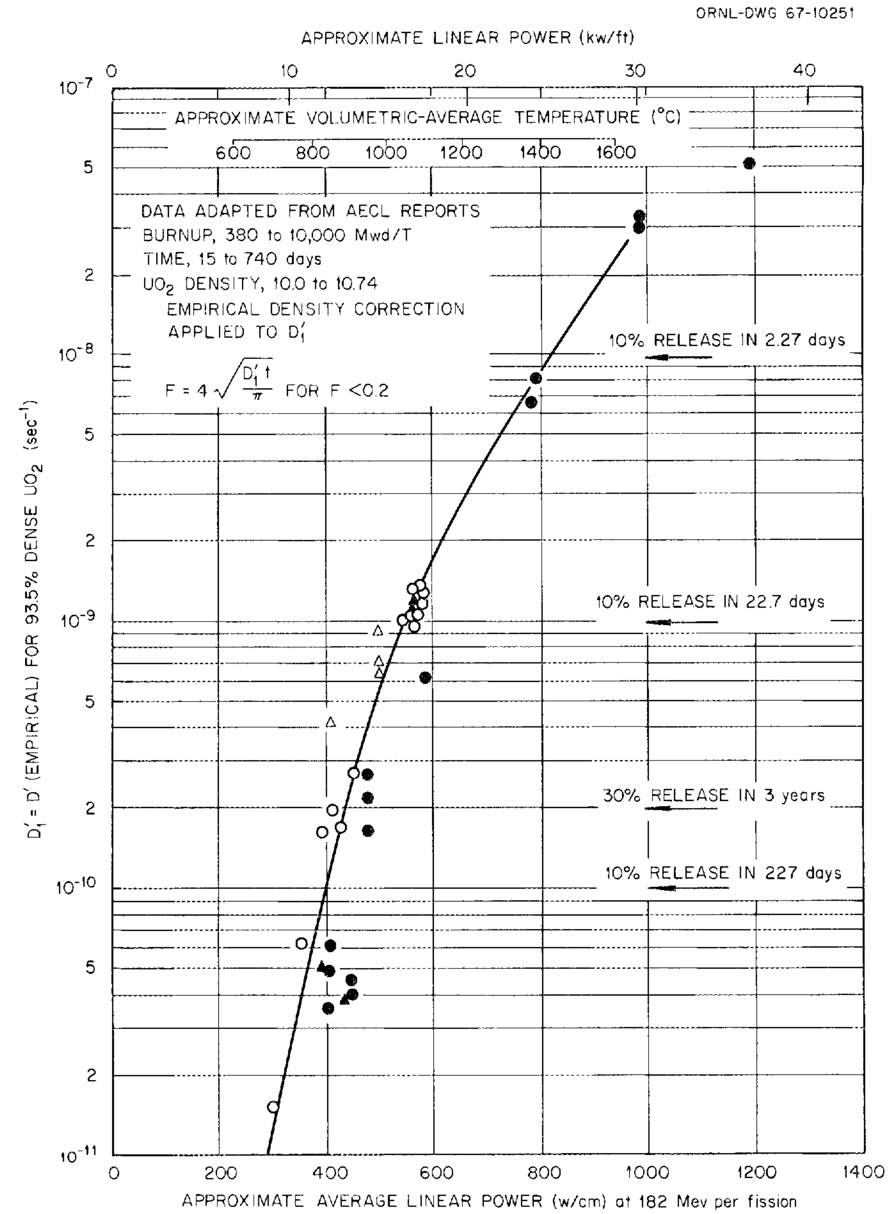

Fig. 1.3. Preliminary Correlation of Stable Fission-Gas Release from UO_2 with Linear Power.

1.3 BEHAVIOR OF FISSION PRODUCTS DURING IN-PILE DESTRUCTION OF REACTOR FUELS

(AEC Activity 04 60 10 01 1)

S. H. Freid B. F. Roberts

Evaluation and interpretation of the results of the first two containment vessel experiments, previously reported,[1,2] has been delayed because of incomplete analytical results. The iodine data from the May packs used in the second experiment are relatively complete and are given in Table 1.2 and plotted as a function of time in Fig. 1.4. The corresponding information for experiment 1 was reported previously.[1] The second experiment differed from the first only in the temperature in the containment vessel, 103°C versus 40°C. In both cases dry air was present in the

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[1]C. E. Miller, Jr., et al., ORNL Nuclear Safety Research and Development Program Bimonthly Report for March–April 1967, USAEC Report ORNL-TM-1864, p. 10, Oak Ridge National Laboratory, May 1967.

[2]C. E. Miller, Jr., S. H. Fried, and B. F. Roberts, ORNL Nuclear Safety Research and Development Program Bimonthly Report for May–June 1967, USAEC Report ORNL-TM-1913, p. 8, Oak Ridge National Laboratory, July 1967.

Table 1.2. Iodine Distribution in May-Pack Components After Second In-Pile UO_2 Meltdown Experiment

May-pack sampler number	1	2	3	4	5
Time sample taken, hr[a]	0.95	3.12	6.15	7.57	10.35
Iodine distribution,[b] $(A/A_0) \times 100$					
Absolute filter (F-700)	4.63	2.15	0.49	0.72	0.56
Silver screens	2.47	5.09	10.12	5.78	9.09
Charcoal filter papers	2.34	3.22	4.11	7.63	12.87
Charcoal filter beds	2.42	2.23	2.33	3.16	2.52

[a]Time after containment vessel was removed from the reactor.

[b]A = iodine activity on component
A_0 = total iodine activity in May packs at t_0 (from extrapolation).

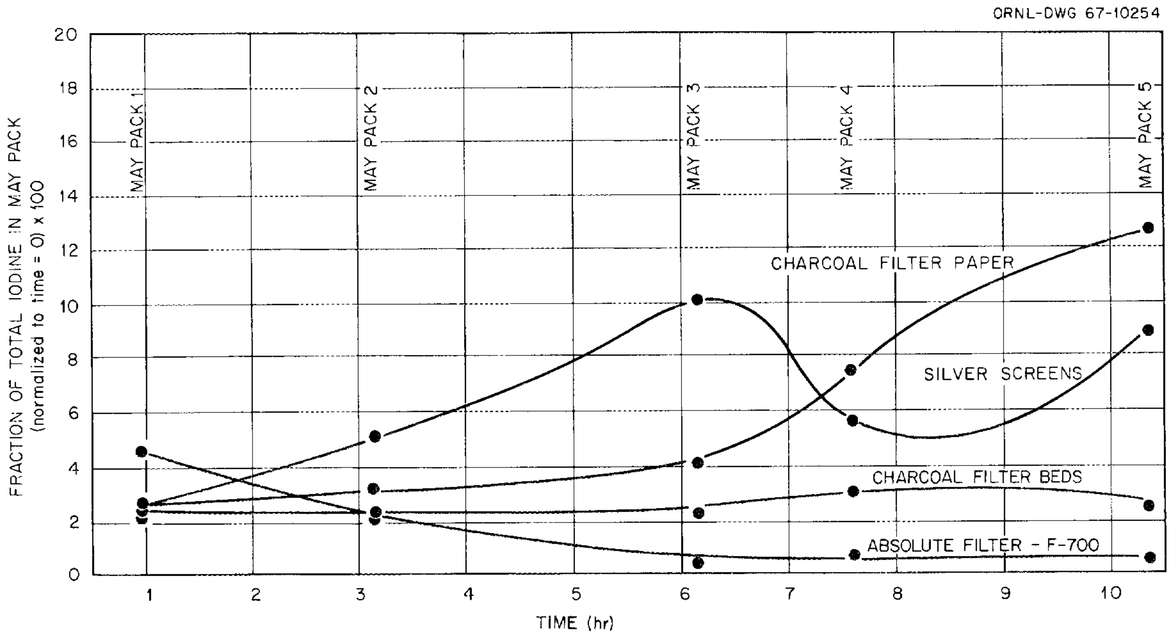

Fig. 1.4. Iodine Distribution in May-Pack Components After Second In-Pile UO_2 Meltdown Experiment.

furnace area and in the containment vessel. These results indicate that the iodine species desorb from the hot (103°C) containment vessel surfaces and change to a more penetrating form upon aging. Some modification of the containment vessel has been necessary during this period to separate the secondary from the primary off-gas systems.

2. WATER REACTOR PROGRAM — ENGINEERING SAFEGUARDS

2.1 MODEL TESTS OF NATURAL TRANSPORT PHENOMENA (AEC Activity 04 60 80 01 1)

L. F. Parsly J. K. Franzreb
J. L. Wantland

The Nuclear Safety Pilot Plant (NSPP) is a facility for studying the behavior of fission products in a containment vessel on a pilot-plant scale. A 10-ft-diam by 15-ft-high model containment vessel is used for the studies, which include studies of both natural transport effects and countermeasures.

The principal effort during the current reporting period has been on the spray test experiments, which are reported separately. Work on this program has included completion of the analytical chemistry and data reduction for NSPP run 19 and further development of a computer program for analysis of thermal convection currents in a closed cylindrical bank.

NSPP run 19 was an experiment in which elemental iodine was released into dry air at room temperature in the model containment vessel. The vessel atmosphere was stirred continuously by means of an electric fan (3000-cfm rating). The time-dependent behavior of the iodine was studied over a period of 70 hr. Table 2.1 presents the data collected by the three May-pack clusters used in the run. Eleven of the May packs were used to collect samples, and one (No. 4) was left as a blank. For this May pack, the number printed as "concentration" is the total activity collected. The principal significance of the blank sample data in this run is that, except for the impregnated charcoal beds (which collect almost nothing) and the check valves, the activity on the components of the May packs that took samples exceeds that on the corresponding blank sampler components by a factor of 10 or more. This means that even in the later stages of the experiment, the values reported are significantly above background.

The data from this run appear to indicate that significant methyl iodide formation occurred. The data for the concentration of the iodine fraction collected by the charcoal beds in the May packs are plotted versus time in Fig. 2.1. Presumably the iodine fraction collected is organic

Table 2.1. Data Collected by May Packs in NSPP Run 19[a]

NSPP RUN NO. 19 IODINE RELEASED INTO DRY MCV FAN ON THROUGHOUT RUN
ACTIVITY IN BLANK SAMPLE CHECK VALVE HAS BEEN SUBTRACTED FROM THAT OF OTHER CHECK VALVES

TUBE NUMBER	1	2	3	4	5	6	7	8	9	10	11	12
TIME SAMPLE TAKEN,[illegible]	.25	.75	1.25	BLANK	2.13	3.13	4.13	6.??	9.??	12.??	18.??	23.50
MAY PACK CLUSTER [illegible] ELEVATION 5 FT ABOVE CENTER LINE RADIUS = 40 INCHES												
SAMPLE VOLUME,LITERS	1?.?3	1.7	1.7?	?.?	16.4?	15.9?	15.9?	33.75	32.1?	64.2?	62.40	6?.18
CONC. (DIS/(MIN)(L))	2.24E 07	1.48E 07	1.36E 07	4.26E 06	1.?E 07	7.17E 06	5.?3E 06	2.?7E 06	1.49E 06	?.54E 05	5.44E 05	4.25E 05
PERCENTAGE DISTRIBUTION												
CHECK VALVES	2?.?7?	22.4782	22.2?65	??.?61?	14.5???	2?.4577	14.5454	14.244?	2?.3627	11.6475	8.6436	4.4452
ABSOLUTE FILTER	1?.?3?	24.??-1	25.2921	[illegible]	2?.?633	21.9946	25.4?1?	25.9562	24.3592	1?.??31	11.7760	7.4837
SILVER-PLATED SCREENS	47.47??	45.1?21	4?.1?96	.?383	52.?151	43.2961	42.5134	37.3457	1?.??5?	17.344?	5.4044	1.6021
FILTER TUBE	7.?227	7.?342	11.?37	.1341	?.5?	1?.??6?	1?.4269	7.8455	5.9677	4.?263	?.9795	1.2192
PLAIN CHARCOAL BEDS	?.??17	?.3??	1.1637	.1112	2.?943	4.1477	?.?955	12.9962	27.73??	45.1561	73.1893	??.2309
IMPREG. CHARCOAL BEDS	?.??	?.?717	?.??34	?.?25	?.?45	?.?59	?.?37?	1.6373	3.7156	?.?232	?.?074	0.0190

MAY PACK CLUSTER [illegible] ELEVATION AT CENTER LINE RADIUS = 31 INCHES NORTH												
SAMPLE VOLUME,LITERS	12.?	12.11	11.3?	?.?	17.7?	17.1?	17.1?	33.??	34.2?	64.2?	69.??	69.40
CONC. (DIS/(MIN)(L))	2.4?E ?7	1.57E 07	1.5?E 07	4.43E 06	1.??E 07	7.74E 06	5.22E 06	1.7?E 06	1.42E 06	?.38E 05	5.53E 05	5.21E 05
PERCENTAGE DISTRIBUTION												
CHECK VALVES	2?.?6?3	21.7?1?	14.2351	9?.12??	5.??63	7.1474	6.?981	28.1743	2?.1?9?	4.5425	6.5963	9.0523
ABSOLUTE FILTER	1?.???	17.6?7?	22.?671	?.6333	21.445?	25.4?33	23.6?1?	26.?253	?.5?52	1?.?376	11.?547	7.3561
SILVER-PLATED SCREENS	57.?135	4?.?195	5?.6?61	?.?244	57.4337	56.5452	53.228?	29.5153	31.3495	26.5?8?	8.2605	2.1790
FILTER TUBE	?.2??	?.?141	1?.?24?	?.?337	?.6??	?.1596	9.7557	5.5?44	7.5454	6.?593	4.1835	3.1092
PLAIN CHARCOAL BEDS	?.??	?.?625	1.3417	.??95	6.1?72	2.6654	6.7969	9.7955	31.4829	43.?46?	69.7892	78.2783
IMPREG. CHARCOAL BEDS	?.?1?	.?14?	?.?51	.2?93	?.??53	?.??1	?.4495	?.??53	?.?9??	?.??54	?.0158	?.0153

MAY PACK CLUSTER NO. 6 ELEVATION [illegible] FT [illegible] CENTER LINE RADIUS = 40 INCHES												
SAMPLE VOLUME,LITERS	?.?	?.3?	?.1?	.?	13.6?	13.5?	14.1?	23.4?	27.??	52.8?	57.6?	57.48
CONC. (DIS/(MIN)(L))	2.4?E 07	1.33E 07	1.4?E 07	4.6?E 06	1.1?E 07	?.37E 06	5.57E 06	2.24E 06	1.67E 06	1.16E 06	7.2?E 05	5.18E 05
PERCENTAGE DISTRIBUTION												
CHECK VALVES	?.?	?.?6?	?.?267	9?.4?22	5.1?27	6.9251	7.92?2	5.9185	5.6761	4.3611	6.595?	7.9295
ABSOLUTE FILTER	2?.??	2?.??3	2?.7134	?.?634	21.7?1?	27.76?1	36.?55?	2?.524?	?2.121?	2?.61??	11.?92?	7.2538
SILVER-PLATED SCREENS	7?.??41	?4.44??	53.??42	?.1594	?.7?8?	50.7651	36.4997	43.8198	26.1409	1?.?638	4.3261	2.2915
FILTER TUBE	?.?415	?.?3?3	1?.3753	.?279	?.??	11.?333	12.2362	7.8679	8.?615	7.2416	4.5342	2.4451
PLAIN CHARCOAL BEDS	.??9?	1.?2?1	1.?654	.1269	2.?6?5	4.11?5	6.4813	12.9641	27.3859	48.2159	71.9487	8?.?387
IMPREG. CHARCOAL BEDS	?.?	?.?3?	?.?4?	?.2141	?.?13	?.?916	?.5387	?.?26?	?.?139	?.?938	?.1149	?.0405

[a]Values are good to only two significant figures.

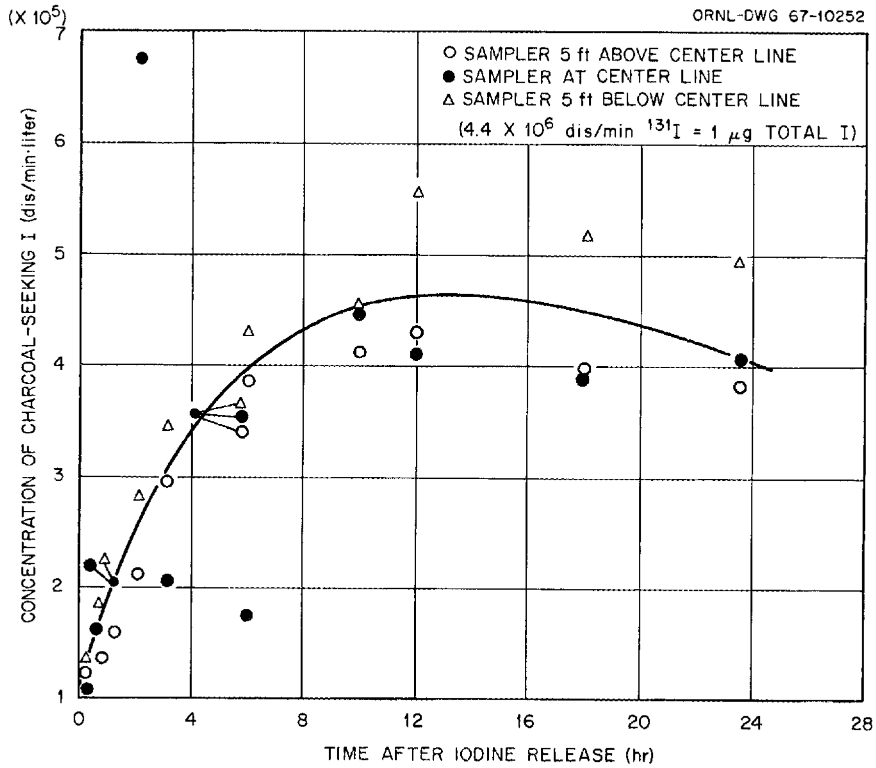

Fig. 2.1. NSPP Run 19 — Fraction of Charcoal-Seeking Iodine Versus Time.

iodide. The concentration increased by a factor of 4.5 during the first 12 hr and then started to decrease. The methyl iodide formed amounted to about 2% of the iodine present in the MCV atmosphere at time zero or about 1% of the iodine supplied for the experiment.

In Fig. 2.2 we have plotted the normalized concentration of iodine (C/C_0), excluding the charcoal-seeking fraction, versus time for run 19 and for run 18, which was similar except that forced circulation was imposed for the first 10 min only. The run 19 data fit the model of reversible adsorption with partial wall coverage proposed by Watson, Perez, and Fontana[1] using $k_c = 6 \times 10^{-3}$ cm/sec and $\alpha = 0.004$ in their Eq. (7).

The run 18 behavior is more complex; it does not appear to fit the above model, nor does it fit a molecular diffusion model.

The computer program was briefly described in the preceding report in this series.[2] During the present reporting period we decided to organize the program as a series of subroutines called by a relatively short master program. This will facilitate compiling the several sections of the calculation and allow the critical sections to be optimized more readily. The calculation subroutines have all been written and compiled and some short test cases have been run. Some input and output routines still have to be written.

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[1]G. M. Watson, R. B. Perez, and M. H. Fontana, Effects of Containment Size on Fission Product Behavior, USAEC Report ORNL-4033, Oak Ridge National Laboratory, January 1967.

[2]L. F. Parsly, J. L. Wantland, and J. K. Franzreb, ORNL Nuclear Safety Research and Development Program Bimonthly Report for May–June 1967, USAEC Report ORNL-TM-1913, p. 18, Oak Ridge National Laboratory, July 1967.

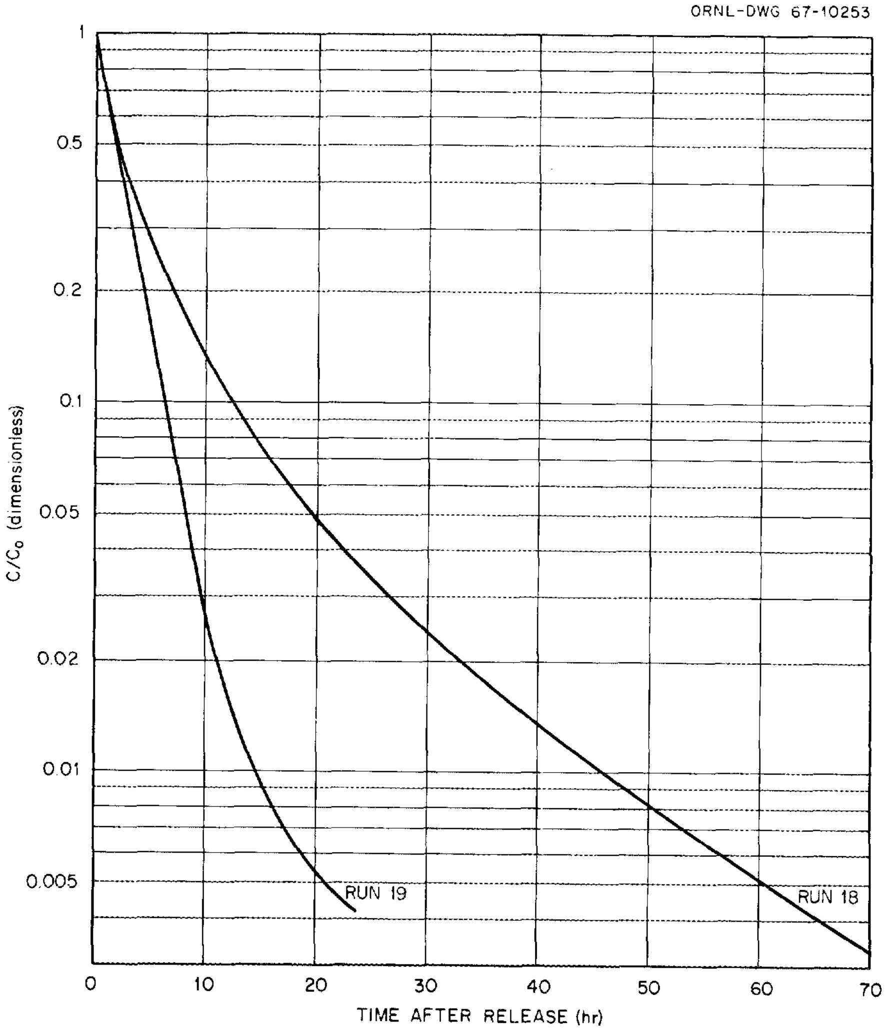

Fig. 2.2. Behavior of Iodine, Excluding Fraction Collected by Charcoal, Versus Time in NSPP Runs 18 and 19.

2.2 SPRAY AND POOL-SUPPRESSION TECHNOLOGY PROGRAM (AEC Activity 04 60 80 01 1)

T. H. Row, Coordinator

The spray and pool-suppression technology program now includes work on literature reviews, a survey of additives for possible use as sprays, corrosion tests, single-dropwind-tunnel studies, scale-model engineering tests, a study on removal of fission products in a pressure-suppression type of containment, and radiation stability of solutions. Close liaison is maintained with the nuclear industry so that the program covers the practical engineering aspects of the application of spray technology to nuclear facilities.

Behavior of Fission Products in Gas-Liquid Systems (B. A. Soldano, W. T. Ward)

The use of additives to improve the efficiency of aqueous spray systems in removing iodine and related iodides raises the question of what effect pH has on this process. A single-drop wind-tunnel study was completed in which the effects of pH variation on both the mass transport coefficients and the distribution coefficients of CH_3I into a 1 wt % solution of $Na_2S_2O_3$ were examined.

The single-drop experimental procedures and method of data analysis remain identical to those described in the May—June bimonthly report, with the drop radius being fixed as before at 0.21 cm. The sole variable in this study was pH. The solution pH was adjusted by the addition of 0.3 wt % H_3BO_3 to the 1% $Na_2S_2O_3$ solution to give a pH of 5.8. Further acidification to a pH of 4.0 was achieved by the addition of HCl. The result obtained at a pH of 6.1 with 1% $Na_2S_2O_3$ was reported in the previous bimonthly report. The basicity variations were made by the addition of NaOH to the $Na_2S_2O_3$—H_3BO_3 solution.

The distribution coefficients, K_d, of CH_3I between the aqueous and gas phases were calculated as follows:

$$K_d = \left(\frac{\text{millimoles } CH_3I}{\text{per cc solution}}\right) \Bigg/ \left(\frac{\text{millimoles } CH_3I}{\text{per cc gas}}\right) .$$

Both the kinetic and distribution effects of pH on the adsorption of CH_3I are shown in Figs. 2.3 and 2.4. It may be noted on Fig. 2.3 that the mass transport coefficient, v_t, of CH_3I rises sharply as the system is made basic. The maximum v_t value at pH 7 is about 9 times higher than the minimum obtained at pH 5.8.

Significant insight into one of the primary processes controlling the mass transport of CH_3I into water is obtained by a comparison of the v_t values at 26 and 40°C and a subsequent calculation of the energy of activation required for the process. The experimental fact is that the energy of activation at a pH of 7 and higher is approximately zero; this reflects a temperature insensitivity of the CH_3I mass transport process. In the acidic range, however, the v_t values increase with temperature. The energy of activation on the acidic side is approximately 10,000 cal based on measurements at four temperatures (26, 34, 40, and 47°C) at a pH of 5.8. No detailed measurements were performed at a pH of 4.0 because the solution is unstable.

It has been postulated by Franck and Haber[1] that the primary step in the photolysis of CH_3I in water is the ionization of water. Since acidification inhibits the ionization of H_2O, an activation energy of 10,000 cal (comparable to the electrical free energy required to ionize water) would be expected if the Franck mechanism is involved. On the basic side, the water is already ionized, so no additional energy is required to make the photochemical reaction proceed (temperature insensitivity being a common characteristic of this type of reaction).

A direct test of the role of the photon was conducted by measuring <u>in the absence of light</u> the mass transfer of CH_3I in 1% $Na_2S_2O_3$ at pH 7.3, and $T \cong 26°C$. The resultant value of 1.73×10^{-2} cm/sec (Fig. 2.3) is definitely lower than the mass transfer coefficient obtained under identical conditions, except for the existence of a fluorescent light source.

The study of pH dependency thus leads to the conclusion that photolysis, as well as pH, may well play a role in the process of alkyl halide mass transfer into water. Moreover, since the pH dependence of the

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[1] J. Franck and F. Haber, Sitz-Preuss Akad. Wiss., p. 250, 1931.

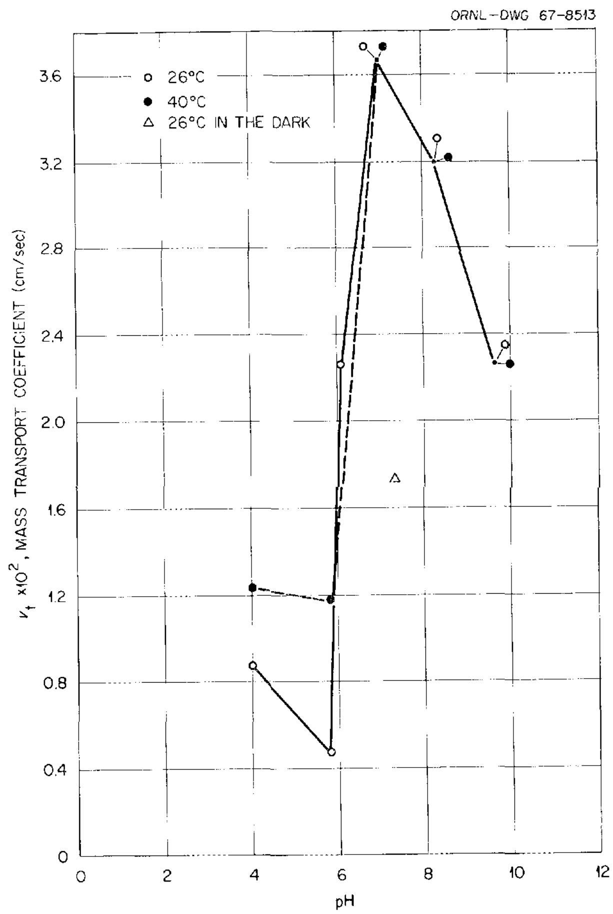

Fig. 2.3. Effect of pH and Temperature on the Mass Transport of CH_3I.

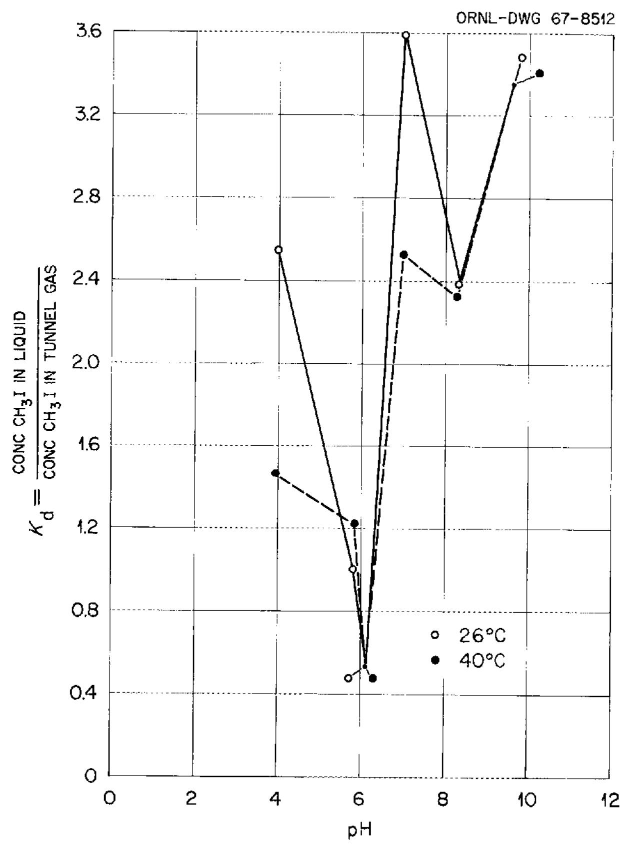

Fig. 2.4. Distribution of Methyl Iodide as a Function of pH and Temperature.

equilibrium distribution coefficients (Fig. 2.4) also roughly parallels that of the mass transport coefficients, from both a kinetic and thermodynamic standpoint, it appears that a pH of 7 or higher provides the most suitable environment for absorbing CH_3I. Since these effects are large, it appears that buffering the spray system to insure alkalinity may well be advisable.

Spray Program in the Nuclear Safety Pilot Plant
(L. F. Parsly, J. K. Franzreb)

Analytical Studies. A thorough review of the chemical engineering literature applicable to the spray program was made, and a memorandum summarizing the salient features was drafted. The study confirms the conclusion that absorption of I_2 into reagents is gas-film controlled (we calculated that this would be true for any I_2 concentration below 20,000 mg/m^3 if 1% sodium thiosulfate solution were used). Likewise, because of a highly favorable distribution coefficient, absorption of HI into water is gas-film controlled (at 25°C the distribution coefficient for HI is 9.2×10^6).

On the other hand, the data for methyl iodide absorption indicate that liquid-phase diffusion and reaction kinetics are involved in determining its removal by sprays; the reported rates are significantly less than those expected for a gas-film controlled reaction but substantially higher than expected for transfer into a rigid drop in the liquid phase. It appears that both circulation of the liquid inside the drop and chemical reaction are contributing to the increase above the rigid drop rate.

Experimental. Three runs (NSPP runs 20—22) were made to test removal of elemental iodine from the model containment vessel atmosphere by sprays. The first two used 0.5 wt % $Na_2S_2O_3$ and 1.4% H_3BO_3 solution, while the third used 0.9 wt % $Na_2S_2O_3$, 1.4% H_3BO_3, and 0.6 wt % NaOH.

All three runs were made at room temperature and approximately 3 psig with a spray header containing 12 Spraco J-140D jewelled-orifice misting nozzles. These have 0.04-in.-diam orifices and a rated delivery of 2.5 gph each at 50 psi or 3.4 gph at 100 psi. The manufacturer reported that these give a mean drop diameter of 100 μ.

In run 20, we succeeded in vaporizing only 10% of the available iodine into the MCV. In our judgment, this introduces a number of uncertainties in the results, and this run should be discarded.

In both runs 21 and 22 we have been able to get excellent checks on the half-time for iodine removal between data on the decay of gamma-ray activity in the MCV and the buildup of gamma-ray activity in the spray solution, which was collected in a pool in the bottom of the model containment vessel. The pool water was continuously circulated past a scintillation counter. The half-time was 34 ± 2 sec. Figure 2.5 presents typical data.

This half-time is substantially greater than that predicted for 100 μ drops at the flow rate used in these runs (2.5 sec). A possible explanation is that the true drop size is larger than the reported value. The observed half-time is consistent with a drop diameter of 275 μ.

The high-pressure spray solution holdup tank required for carrying out spray experiments at elevated temperature and pressure is now being installed. Experiments with nozzles of the type being proposed for containment building spray systems and covering a range of pressures and temperatures will start about September 1.

Radiolysis Studies (H. E. Zittel)

A study of the effects of high-energy radiation on various proposed spray solutions was initiated. To date, the effects of ^{60}Co gamma radiation on the various components of the $Na_2S_2O_3$-boric acid system have been under study. The radiolytic effects are being followed by (1) change in I_2 equivalence of the irradiated solutions, (2) change in pH, (3) measurement of radiolytic H_2, and (4) identification and measurement of radiolytic solids. The effect of change in dose rate and total dose on these parameters is being established. The data obtained to date are presented in Table 2.2.

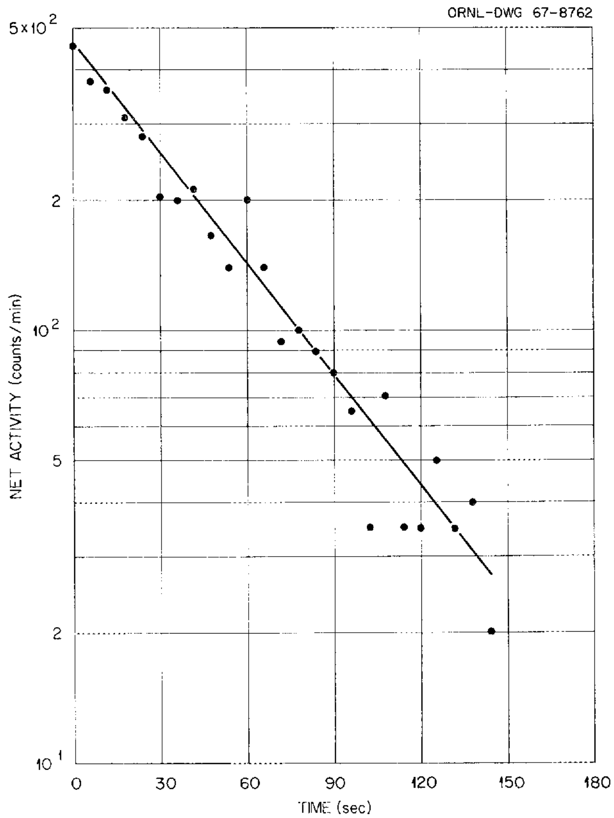

Fig. 2.5. NSPP Run 22 – Removal of Iodine from MCV Based on MCV Gamma-Ray Data.

Table 2.2. Preliminary Results of Radiolysis Studies

Solution No.	Solution Makeup	Dose (r)	pH After Irradiation	ΔI_2 (I_2 equivalence per liter)[a]	G (I_2 equivalence)	Equivalence Destroyed (%)	Solids
1	1 wt % $Na_2S_2O_3$,	9×10^5	9.2	-6.0×10^{-5}		−0.1	None detected
	3000 ppm B,	1×10^7	9.1	-1.7×10^{-3}		−2.6	None detected
	0.153 m NaOH,	2×10^7	9.0	-1.3×10^{-3}		−2.1	None detected
	pH 9.2	5×10^7	8.9	$+9.0 \times 10^{-3}$	0.2	+13.9	None detected
2	3000 ppm B,	9×10^5	9.3				
	0.153 m NaOH,	1×10^7	9.3				
	pH 9.3	2×10^7	9.2				
		5×10^7	9.2				
3	Distilled H_2O,	5×10^5	6.2				
	pH 6.1	1×10^7	6.1				
		2×10^7	6.0				
		5×10^7	5.9				
4	3000 ppm B, pH	9×10^5	4.7				
	4.6	1×10^7	4.7				
		2×10^7	4.6				
		5×10^7	4.5				
5	1 wt % $Na_2S_2O_3$,	9×10^5	4.4	$+1.2 \times 10^{-3}$	1.3	+1.8	None detected
	3000 ppm B, pH	1×10^7	3.8	$+8.0 \times 10^{-3}$	0.8	+12.4	None detected
	4.9	2×10^7	3.7	$+15.7 \times 10^{-3}$	0.8	+24.3	Solids present
		5×10^7	3.6	$+36.5 \times 10^{-3}$	0.7	+56.2	Data not available at present time

[a] A negative value indicates an increase of I_2 equivalence on radiation.

2.3 HIGH-EFFICIENCY AIR FILTER ENGINEERING MANUAL (AEC Activity 04 60 80 01 1)

C. A. Burchsted A. B. Fuller

The high-efficiency air filter engineering manual is being prepared to provide a guide for engineers and architects to the requirements for and design of air-filter systems that require the use of HEPA filters. Chapters 1 through 5 have been issued for review and some revisions have been made. However, response to the review request has been disappointing; no written comments have yet been received from the AEC. Preparation of the remaining chapters continued.

2.4 CHARACTERIZATION AND RELEASE TECHNOLOGY (AEC Activity 04 60 80 01 1)

R. E. Adams

Investigations are being made of the characteristics of those fission products likely to be released in nuclear accidents, and their behavior and methods for control are being studied under the various conditions that would be expected in accidents in water-cooled reactors.

Removal of Iodine by Solid Sorbents (R. D. Ackley, Z. Combs, F. V. Hensley)

Current emphasis is on the trapping of elemental radioiodine and radioactive methyl iodide by commercial iodized charcoals. Activities during this period included (1) determination of I_2 removal efficiencies for iodized charcoals at high loading and high relative humidity, (2) continued investigation of loss of impregnant by iodized charcoals at elevated temperatures, (3) additional $CH_3{}^{131}I$ removal tests on samples of "weathered" charcoal, and (4) testing of impregnated silica gel in connection with an effort to develop a noncombustible agent for $CH_3{}^{131}I$ trapping.

Because iodized charcoal contains one or more iodine-containing substances, the question has been raised as to whether or not the capacity of such charcoal for elemental iodine has been reduced to a low and thereby dangerous level. Accordingly, this question is being investigated and some results have been obtained. In these initial tests, extremely high loadings of I_2 were employed for the reason that if the charcoals performed well at these high loadings, their performance (toward I_2) at lower loadings (of I_2) would be satisfactory. Four tests were completed. The I_2 loadings on the charcoal, as based on the amount of charcoal in a 2-in. depth, were varied from test to test within the range of 3 to 7 wt %. The variation in I_2 removal efficiency within this range of loading was not pronounced. Some of the other pertinent test conditions were 25°C, 40 fpm, approximately 96% relative humidity, and I_2 injection time of 8 to 15 hr followed by continued air sweep for at least 9 hr. Averages of the I_2 removal efficiencies are given in Table 2.3. The first two listed

Table 2.3. Average Iodine Removal Efficiencies of Three Types of Charcoal

Type of Charcoal	Lot No.	I_2 Removal Efficiency (%)	
		For 1-in. Depth	For 2-in. Depth
MSA 85851	93066[a]	99.92	99.95
BC-727	01345	99.93	99.94
Experimental (2%)[b]	4767[a]	99.93	99.95

[a]ORNL assigned lot number.

[b]I_2 loading.

types of charcoal are commercial materials and the third (also commercial in orgin) is similar to BC-727 but iodized to a lower degree of impregnation.

As may be noted, all three charcoals exhibit excellent I_2 trapping capability under the conditions cited. Additional tests will be conducted in which bulk-phase condensation of water will be induced in the charcoal.

Observations made elsewhere showed that iodized charcoals emit impregnant in the form of I_2 upon heating to temperatures well below their ignition temperatures. This may constitute a serious problem because of a possible loss in effectiveness and/or evolution of radioactivity if the charcoal had previously been used to trap radioactive I_2 or CH_3I. Consequently, an experimental program to study the evolution of radioactivity from impregnated charcoals so exposed was initiated. Some initial results were reported in the previous bimonthly report[1] for temperatures of 100, 150, 200, and 260°C. Subsequent work indicates that the quoted temperatures are in error and that the average temperatures should be approximately 85, 135, and 185°C. The 260°C tests, however, were performed in a revised and improved apparatus, and the quoted temperature is correct. Additional tests of this type are being continued, and in one set

[1]R. E. Adams et al., ORNL Nuclear Safety Research and Development Program Bimonthly Report for May—June 1967, USAEC Report ORNL-TM-1913, pp. 9—12, Oak Ridge National Laboratory, July 1967.

of experiments performed to compare the behavior of BC-727 and the experimental charcoal listed in Table 2.3, the latter was observed to be considerably less susceptible to loss of radioactivity on heating. Results for these and other types of charcoal are to be detailed in subsequent reports.

A limited amount of data obtained previously had indicated that subjecting impregnated charcoal to continued humid air flow caused some reduction in $CH_3{}^{131}I$ removal efficiency. In order to investigate this point further, two separate groups of iodized charcoal samples (type MSA 85851) are being weathered. One group is being exposed to ambient air from the Oak Ridge Research Reactor (ORR) building with the approximate operating conditions being 20 fpm, 78°F, and 50% relative humidity. The other group, located in the laboratory, is exposed to clean humidified air; approximate operating conditions are 40 fpm, 78°F, and 50% relative humidity. Periodically individual samples are withdrawn from the weathering setups and tested for $CH_3{}^{131}I$ retention under the following conditions: 25°C, 40 fpm, 65% relative humidity, and approximately 1 mg CH_3I injected per gram of charcoal. Results thus far obtained are summarized in Table 2.4.

The decreases noted in Table 2.4 appear to result from the flow of humid air rather than aging, since tests on unexposed samples of the same batch of charcoal over the time period involved have not as yet revealed a significant effect due to aging. While certain trends are indicated

Table 2.4. $CH_3{}^{131}I$ Removal Efficiency of Weathered Charcoal

Exposure Time (months)	Location	$CH_3{}^{131}I$ Removal Efficiency for a 2-in. Depth (%)
0		99.3
3	ORR building	98.4
6	ORR building	97.1
3	Laboratory	98.9

by the results in Table 2.4, it is probably too early to draw firm conclusions.

Two different grades of high-surface-area silica gel were impregnated with 3 wt % KI + I_2 in a 2:1 molar ratio. These agents tended to evolve I_2 on standing, but fresh samples did exhibit some $CH_3{}^{131}I$ removal capability under dry conditions. In tests at 40 fpm and less than 3% relative humidity and with approximately 0.3 mg CH_3I injected per gram of impregnated silica gel, $CH_3{}^{131}I$ removal efficiencies of approximately 10% were observed for 2-in. depths. Under the more humid conditions of actual interest, even lower efficiencies are anticipated. This particular material therefore does not appear to represent an effective noncombustible trapping agent.

Filtration of Solid Aerosols (R. J. Davis, J. S. Gill, J. Truitt, W. D. Yuille)

Continuation of the measurement of filter efficiencies of high-efficiency filter media with electric-arc-generated stainless steel–UO_2 aerosol has provided the following information.

A series of 12 tests of one filter paper under the same conditions (i.e., aerosol in dry air and at a linear velocity of 5 fpm) was performed to demonstrate precision of the test method. Except for one low-efficiency value, the average was 99.97%, the average deviation from the average was 0.01%, and the extreme deviation was 0.02%. It is indicated that the irrational errors of the method amount to 0.01% on the average. It was decided in future experiments that four measurements of each condition should be made to enable us to confidently discard occasional very low values.

A series of tests was run with each of five different samples of filter media (each of which is claimed to be at least 99.97% efficient with DOP aerosol of particle size 0.3 μ or greater) at linear aerosol velocities of 3.5, 5, 7.5, and 10 fpm, with the test aerosol in a water-saturated atmosphere at room temperature. At the rated linear velocity of 5 fpm, three of the papers were 99.97% efficient and two were 99.92 to 99.93% efficient. Filter efficiencies were not much affected by velocity from

3.5 to 7.5 fpm, but efficiencies were significantly decreased at 10 fpm, namely, to values ranging from 99.95 to 99.12%.

Exploratory tests in which the filter medium was wetted with water resulted in very low, but erratic, filter efficiencies. It is indicated that the previously noted deleterious effect of humidity on filter efficiencies is an effect of liquid water on the filter medium.

A program is being established that will characterize aerosols of interest. An experiment is now being assembled that will enable us to study processes such as coagulation, diffusion, and settling as a function of turbulence, initial aerosol concentration, and humidity.

Size Classification of Submicron Particles (H. Buchholz)

The lower limit of size classification in the cascade impactor, as it is normally used, is particles of 0.5 μ diameter. In nuclear safety research, smaller particles that remain airborne for a long time are important. We have considered the slip of particles under reduced pressure as a way to extend the lower limit of the cascade impactor down to at least 0.01 μ. It was found by calculation that the Andersen sampler, a six-stage cascade impactor, would be adequately efficient in classifying particles in the range 0.01 to 1 μ if an air flow rate of about 8 liters/min and an internal pressure of about 40 mm Hg was maintained. Therefore, experimental work was started with an Andersen sampler to convert it to a low-pressure device with a backup filtration system.

The first verification of the high efficiency for small particles under low pressure was attained with a $CsNO_3$ aerosol tagged with ^{137}Cs. Fifty percent of all deposited material was found on the fourth stage at 40 mm Hg, and only 0.05% penetrated to the backup filter, whereas at 750 mm Hg, 99% of the material passed through the backup filters. This indicated high penetration of the particles under ordinary pressure.

Another cascade impactor (in addition to the Andersen sampler) designed for low-pressure operations was constructed to provide lower pressure drops, greater ease of decontamination, and faster sample plate exchange. Hole diameter and hole numbers were calculated according to a desirable increase in D_{50} (particle diameter of 50% deposition efficiency) by a factor of 3 from stage to stage with particles having a density of 6 g/cm^3.

A particle generator was developed to produce NaCl particles from a nearly saturated atmosphere. This was accomplished by evaporating a solution of NaCl tagged with ^{24}Na onto quartz wool that had the appearance of cotton and heating it at 800°C with air flowing at 400 cm/min. The mass concentration of the particles downstream from the generator was found to be constant within a few percent during several hours. Because this generator produced particles with a constant size distribution and constant number concentration, the effect of varying the sampler operating conditions could be readily determined. The first tests of the samplers using the NaCl particle generator with ^{24}Na tracer disclosed a reasonable material distribution among the stages and on each sample plate.

When the sampler is used to classify particles under simulated reactor accident conditions, adsorbable gaseous fission products such as iodine may be present, as well as particulate fission products. This would interfere with the proper operation of the sampler, since molecular iodine would adsorb on the sample plate by a diffusion mechanism. For sampling particulates, it is important that the particle sampler collect only the matter settling on the sample plates by impaction. Three silver screens placed in the gas stream near the entrance of the first stage were very efficient in reducing the gaseous iodine concentration. The sample plates showed less than 0.7% of the iodine activity collected by the screens, and in another test with NaCl particles, only 1.5% of the particle activity was retained by the screens.

2.5 SEPARATION OF NOBLE GAS FROM AIR USING A PERMSELECTIVE MEMBRANE

(AEC Activity 04 60 80 01 1)

R. H. Rainey

Laboratory experiments are being run to evaluate the use of thin (~1-mil) sheets of dimethyl silicone rubber membrane for the separation of noble gases from air in case of a reactor accident. The permeability of xenon through a 1.75-mil-thick dimethyl silicone rubber membrane was measured by using mixtures of stable xenon and ^{133}Xe tracer and either oxygen or nitrogen (total xenon concentration, less than 0.5%). Contrary to accepted theory, the permeability of the xenon was different when it was mixed with oxygen than when it was mixed with nitrogen. When the xenon was mixed with oxygen, the xenon permeability decreased from about 81 at a pressure drop of 25 psi to about 61 at a pressure drop of 125 psi (Fig. 2.6). Permeability is measured in ml/sec × thickness [cm/area (cm^2)] × [$10^9/\Delta P$ (cm Hg)]. The corresponding permeabilities of xenon in nitrogen were 48 at 25 psi and 46 at 125 psi (Fig. 2.7). The permeabilities of the oxygen and nitrogen in these experiments agreed closely with previously determined values.

Also, in contrast to accepted theory, the permeability of xenon decreased with an increase in product cut (flow rate of gas through the membrane versus flow rate of the gas in the feed stream). For xenon in nitrogen at a pressure drop of 25 psi, the permeability of the xenon decreased from about 82 at a product cut of 10% to 40 at a product cut of 70% (Fig. 2.8). Very similar data were obtained at a pressure drop of 75 psi. The above measurements were made with a 50% product cut.

All previous measurements, including those used in engineering calculations for an advanced water-cooled reactor, were made with a 33 to 50% product cut. Since these values were obtained at approximately the conditions of the conceptual plant, the design and the cost calculations of the plant will not be altered substantially by these new values.

If a permselective membrane is used to separate xenon and krypton from air, the size of the process cascade will largely be determined by the separation of krypton from oxygen. We have proposed that the oxygen

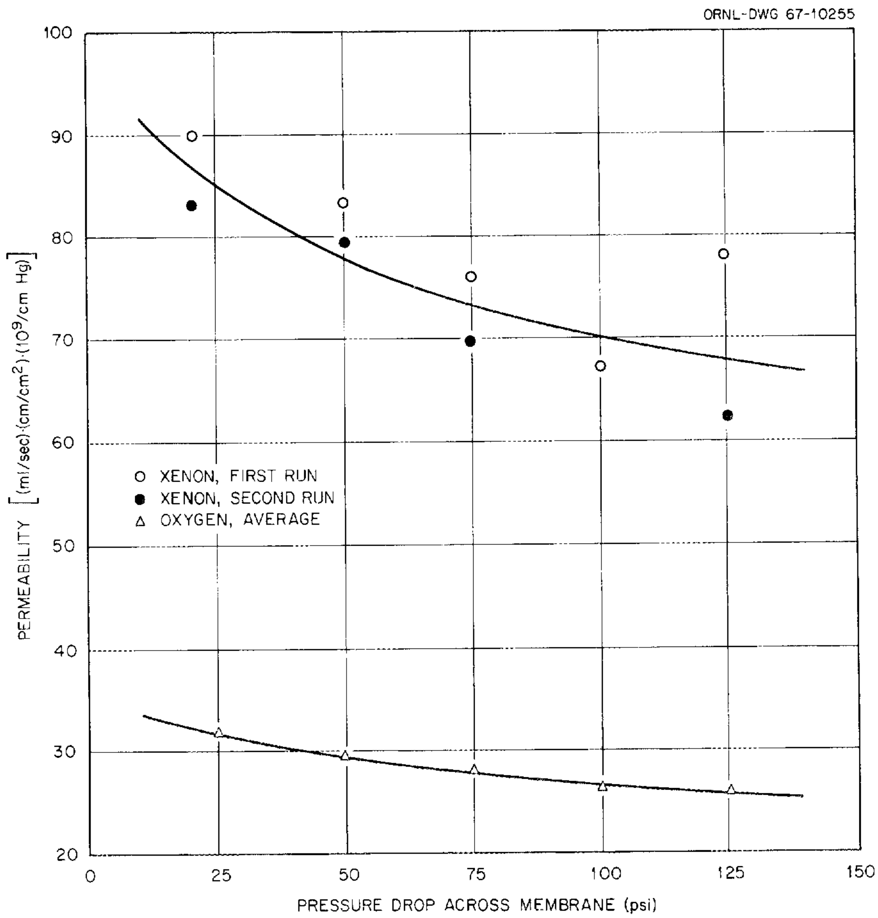

Fig. 2.6. Permeability of Xenon-Oxygen Mixture Through 1.75-mil-Thick Dimethyl Silicone Rubber Membrane.

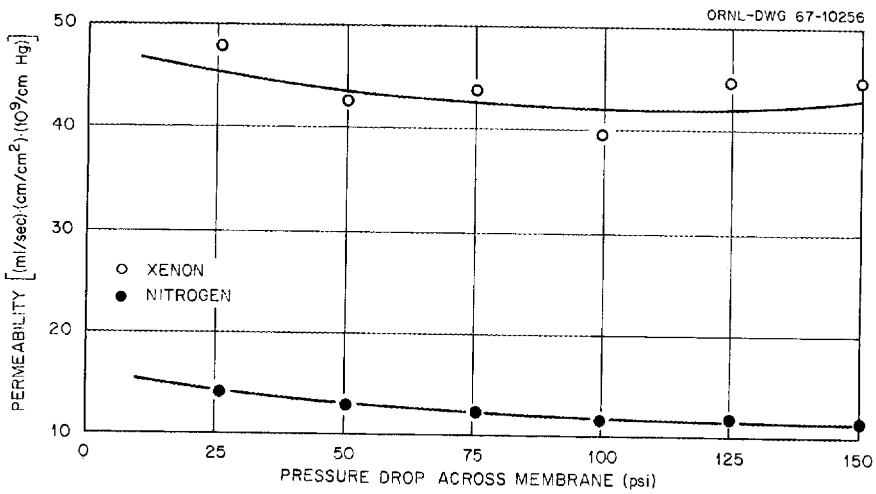

Fig. 2.7. Permeability of Xenon-Nitrogen Mixture Through 1.75-mil-Thick Dimethyl Silicone Rubber Membrane.

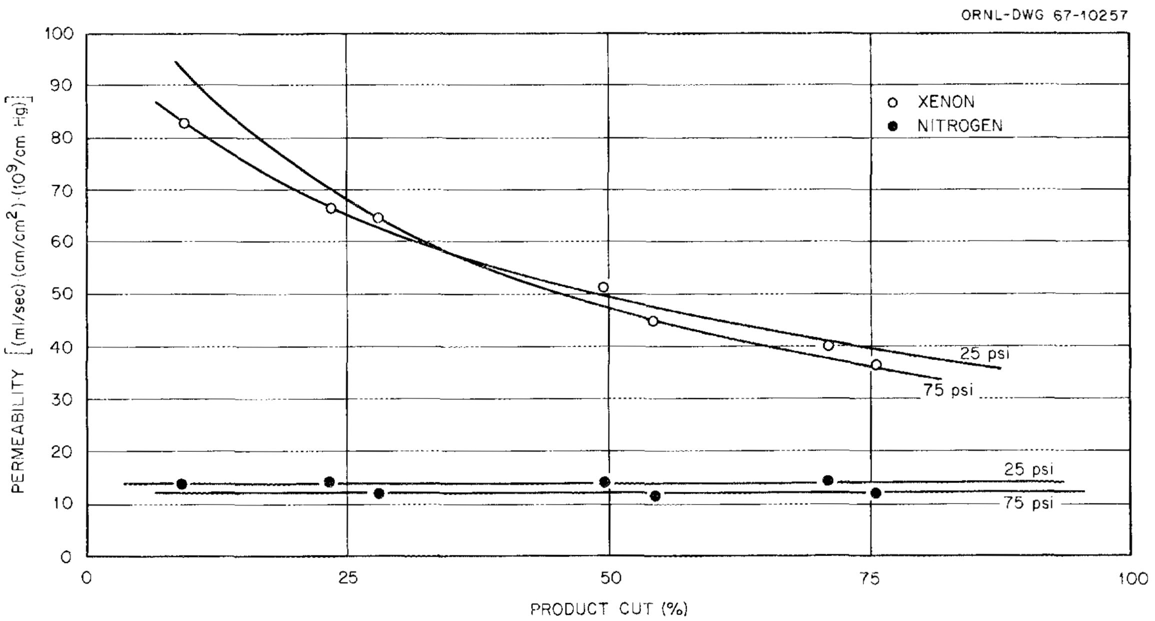

Fig. 2.8. Permeability of Xenon-Nitrogen Mixture Through 1.75-mil-Thick Dimethyl Silicone Rubber Membrane as a Function of Product Cut and Pressure Drop.

be removed by burning either hydrogen or carbon and then removing the water or carbon dioxide products. After this treatment, the argon originally present in the air becomes a major factor in the separation.

The permeability of argon through a dimethyl silicone rubber membrane has been reported to be the same as that for oxygen; these results have been confirmed in our experiments. As previously reported, the permeabilities of all the gases measured decreased slightly with pressure. At a 25-psi pressure difference across the membrane and with a 50% product cut, the permeability of both oxygen and argon were 32.2. At 150 psi, they were 24.5 (Fig. 2.9). The permeability of nitrogen was in agreement with previous measurements: 14.8 at 25 psi and 11.3 at 150 psi.

The thickness of the silicone rubber membrane used in these experiments was measured with a microscope to be 1.75 mils, in agreement with the value reported by GE. A sheet of the membrane that was 2.13 mils thick expanded to 2.84 mils and became brown after being exposed to air saturated with methyl iodide for about a month but did not appear to be otherwise changed.

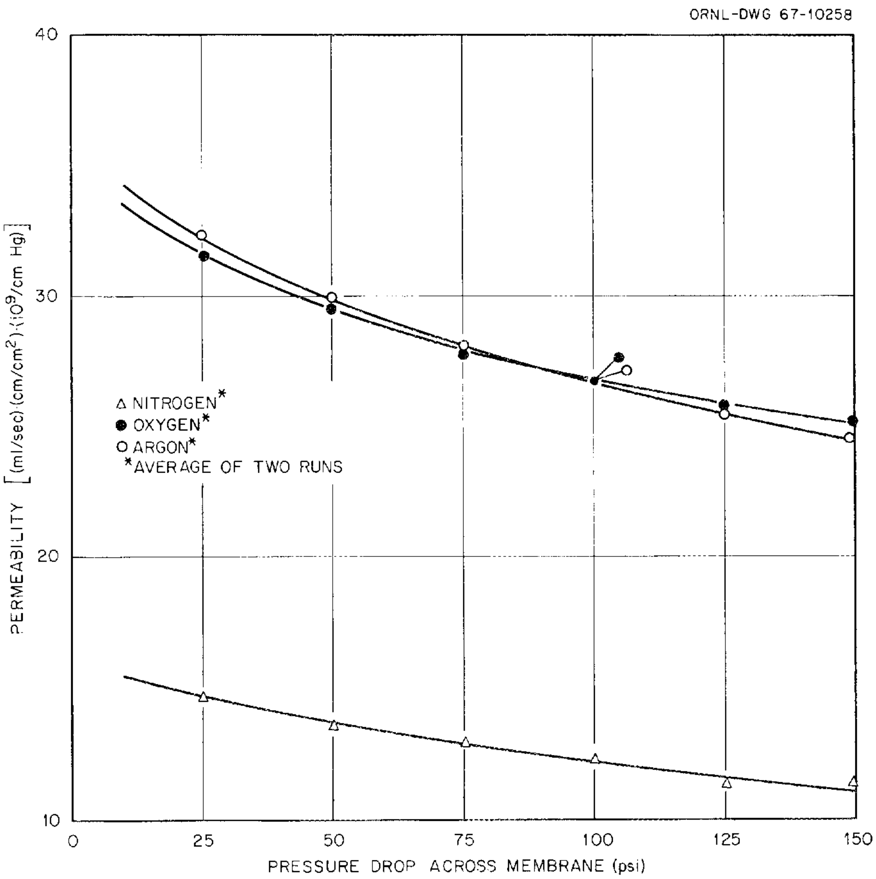

Fig. 2.9. Permeability of Argon, Oxygen, and Nitrogen Through 1.75-mil-Thick Dimethyl Silicone Rubber Membrane.

2.6 ANTISEISMIC DESIGN OF NUCLEAR FACILITIES (AEC Activity 04 60 80 01 1)

R. N. Lyon

ORNL has been assigned the responsibility of a coordinating group to assist the Commission in the planning and execution of a program leading to the development and model testing of antiseismic design for nuclear facilities. The initial approach is to employ a number of architectural firms to attempt to develop a basis for a seismic design technology. The purpose of these studies will be to estimate the ability of current reactor designs to accommodate differential ground displacement and to make a preliminary study of the practical degree to which current designs can be modified with current engineering practice to accommodate more extensive ground displacement.

As of September 1, 1967, subcontracts will be in effect with four engineering concerns to study four concepts for preventing reactor safety system failures due to ground shaking or differential ground displacement at a reactor site. The contract with Holmes and Narver has been in effect for three months and is aimed at determining the cost of engineered substrates between the reactor system foundations and the ground and the benefits and limitations of using such substrates. In the first two months, studies were made of two hypothetical sands of differing density and internal friction. The width of supporting sand bands due to vertical faulting was estimated for the two sands for two fault locations, one at the edge of the 140-ft-diam containment base and one along the diameter. It appears feasible to strengthen containment vessels to resist up to 6-ft vertical differential displacement. An empirical formula obtained by Rowe and Parker (Geotechnique, 1965) was used to approximate the load on a cylindrical wall for the case of a medium dense sand. Present Indian Point II containment walls appear to be capable of withstanding a 6-in. horizontal displacement with the addition of steel to resist both meridional and hoop out-of-plane stress.

Kaiser Engineers plans to investigate the practical limits in unitizing and strengthening reactor systems to resist faulting and shaking.

Daniel, Mann, Johnson, and Mendenhall are studying the cost of supporting the entire plant in water to take care of tsunamis, seiches, and other wave problems. One hull concept involves air cushions to minimize transmission of pressure waves (seaquakes). A major anchoring problem results from the possibility of a large fault under the hull that would cause rapid water flow. Another problem is the large displacement of the hull. Either a very broad hull would be required or a draft of more than the 30-ft depth specified for the offshore case.

Burns and Roe are developing the concept of partial or total support of the reactor containment in a slurry or suspension. They have discussed the problem with drilling-mud manufacturers and with other experts on slurries and non-Newtonian fluids. The problem of maintaining the desired homogeneity, concentration, and physical properties may be more difficult than that of obtaining them initially. A mud with a finite yield strength will have limited leakage and ultimately a zero leak rate at steady state out of the cracks formed by faulting, provided the crack width is not more than two times the yield stress divided by density. Fibers, chopped plastic sheet, and other coarse plugging particles are often added to drilling muds to further reduce the penetration of the muds into cracks.

3. LOFT SUPPORT WORK

3.1 LOFT ASSISTANCE IN OUT-OF-PILE FUEL MELTING EXPERIMENTS (AEC Activity 04 60 50 05 1)

G. W. Parker W. J. Martin

The scope of the ORNL support program for LOFT is being altered to reflect the recent change in LOFT objectives. The major fission-product release mechanism applicable to the new LOFT blowdown model is expected to be the prompt release of cladding-gap activity with cladding failure following depressurization. The release process should then be interrupted by the injection of emergency core cooling water, which will generate sufficient steam to provide the major method of fission-product transport through the pressure vessel. Deposition of fission products from the steam, in addition to the washout from the spray, will reduce the release to the containment. These phenomena therefore become areas of investigation.

The main objectives of the reoriented ORNL LOFT assistance program in fission-product behavior in support of the Emergency Core Coolant Systems Test are (1) to determine in pilot-plant scale the effect of injection of the emergency core coolant on the fission-product release and transport from fuel rods with failed cladding; (2) to study, with irradiated capsules, the parameters that control the release of volatile fission products from the failed cladding and to determine on a larger scale (in the Containment Research Installation, CRI) the degree of retention of released gas activity in the primary vessel; (3) to investigate the effect of burnup on fuel release to provide helpful information for the extrapolation of release results obtained in LOFT from fuel with low burnup to prompt releases from fuel with burnups typical of full-scale power reactors; (4) to determine rate and degree of deposition and retention of radioiodine on stainless steel and painted surfaces and to characterize the time-dependent chemical forms of radioiodine in typical LOFT-type environments; and (5) to evaluate on the CRI scale several LOFT-engineered safeguard devices.

For this investigation, we propose to insert an irradiated UO_2 Zircaloy-clad fuel rod into a center position in the reactor core simulator assembly. The rod will then be heated to a typical "leveling temperature" and finally ruptured rapidly on a steep increase of input power. Coolant sprays will be activated immediately on cladding failure and the heat will be removed by steam formation on the hot rod and core parts. Scoping tests of the above nature would suffice to give early information on effects such as (1) fractional release from a LOFT fuel rod, (2) reduction in the primary vessel, (3) behavior of the resulting aerosol in the containment system, and (4) comparative behavior with different containment surfaces.

Comparative Behavior of Fission-Product Aerosols in a Stainless Steel Containment Vessel Versus a Coated-Surface Containment Vessel

Interpretation of the significance of a change in the LOFT design to include a stainless steel-lined containment vessel instead of the typical reactor coating could be a major benefit of the CRI program. The CRI is especially designed for conducting parallel studies with liners of different materials to determine deposition behavior on stainless steel, carbon steel, concrete, and painted surfaces. Thus, it is likely that complete confidence could be developed in predicting the difference in aerosol behavior that would be encountered if LOFT should later install a special coating over the stainless steel liner.

The CRI was built on a LOFT Assistance Program to meet the need for a larger and more versatile facility for study of the behavior of released fission products under LOFT conditions. The CRI also provides the capability of evaluating the performance of engineered safeguards with realistic fission-product aerosols. The CRI internal recycle filter system permits realistic testing of filters and iodine trapping systems. Provision is also made for the evaluation of pressure-reduction chemical sprays for the removal of various forms of radioiodine. The readily removable liners in the CRI containment vessel permit a direct comparison of the containment-surface environmental effect on fission products between stainless steel and typical coated surfaces. This comparison should be especially important to the new LOFT program.

Fission-Product Release at LOFT Conditions

Parametric studies of release by diffusion should be performed in a series of carefully controlled experiments with the grades of UO_2 in use in power reactors over the fuel-density range of 93 to 95% of theoretical. The effect of burnups up to 20,000 Mwd/T should also be investigated. The initial experiments will be conducted at low burnups typical of LOFT, and later the burnups studied will be extended to cover the present range of water reactor practice.

The basic experiments needed to add understanding and interpretation to the mass of inconsistent UO_2-fission product diffusion release data should include parametric studies of

1. the effect of UO_2 density (93, 94, and 95% of theoretical),
2. the effect of burnup to 50,000 Mwd on average-density UO_2 (94%),
3. the effect of heat ratings to about 25 kw/ft,
4. the equivalent release rate (not diffusion) at grain growth temperatures up to 2700°C.

The proposed diffusion experiments are applicable to the general Nuclear Safety Program. Specifically, however, the calculation of cladding-gap activity, which is fundamental to LOFT, is so dependent upon diffusion release values that a consistent series of well-characterized basic measurements is required before any level of confidence can be expected in a proposed model for gap activity estimation.

We propose to extend the parametric diffusion release studies to include the grain-growth region in order to determine a release rate for both regimes (equiaxial and columnar) equivalent to the diffusion parameter.

Cladding Interaction and Retention in Ruptured Fuel Pins

The study of cladding interaction and retention of fission products is intended to follow up the prompt release test of overrated fuel capsules (see Sect. 1.2 of this report). The difference in the two series is mainly that in the prompt release program the effect of the heat rating of the fuel will be systematically studied, whereas in this series, scoping tests will be conducted of a number of parameters, including the burnup level, the reaction of fission products with the cladding, the result of interreaction of fuel with cladding oxides, and the effect of the interaction on fission-product release and transport.

3.2 CHARACTERIZATION OF FISSION PRODUCTS UNDER LOFT CONDITIONS (AEC Activity 04 60 50 05 1)

R. E. Adams   Ruth Slusher
R. L. Bennett   W. H. Hinds

Iodine-sampling devices are being developed and tested for use in the environments expected in the LOFT experiments. A major problem in the development of a LOFT May-pack sampler suitable for the separation of particulate iodine, elemental iodine, and organic iodides has been the difficulty in defining the proper sequence of materials that will reproducibly separate the particulate iodine from elemental iodine. When high-efficiency filters were used in the initial section to remove particles and particulate iodine, significant quantities of elemental iodine were also retained by these materials. The reverse arrangement, in which the silver screens (for elemental iodine retention) were placed before the high-efficiency filters, resulted in high retention of particles along with the desired elemental iodine removal.

May-Pack Sampler with Initial Honeycomb Section

Recent tests with silver-plated aluminum honeycomb sections as an aerosol—elemental iodine separator have shown promise. As the iodine passes through the channels of the honeycomb it is removed as a consequence of its larger diffusivity, while the larger aerosol particles, with much smaller diffusion coefficients, pass through. Previous tests showed that 99.7% of the elemental iodine introduced was removed by a 5-cm length of silvered honeycomb at 90°C and 90% relative humidity. Only 2.3% of a radioactive aerosol generated by arc-melting an irradiated stainless steel tube with a UO_2 insert was removed by the honeycomb section from a 15% relative humidity air stream. Additional tests in the aerosol facility were completed at higher humidities and the results, to date, listed in Table 3.1, confirm the favorable low collection of the aerosol particles.

Because the silver honeycomb presents a larger surface area to methyl iodide than the previously tested silver screens (for iodine removal), a test was conducted to determine the adsorption characteristics of the

Table 3.1. Retention of Aerosol on Honeycomb at High Humidity

Temperature: 25°C
Flow rate: 1 liter/min

Test No.	Relative Humidity (%)	Retention on Honeycomb (%)
HC1	15	2.30
HC2	91	0.71
HC3	95	0.73

iodide on the honeycomb. The results from duplicate characterization packs are summarized in Table 3.2. Although the retention is greater on the honeycomb than was observed on the eight silver screens (about 0.05%) it is still satisfactorily small.

Formation of Organic Iodides

The study of the formation of volatile organic iodides by the reaction of elemental iodine with the six-coat Amercoat 66 paint system (commonly used for coating steel surfaces) was continued. The coating consists of one coat of Amercoat 64 primer, three coats of Amercoat 66 epoxy, and two coats of Amercoat 66 seal gloss. The reaction was investigated by painting 1.2-liter stainless steel vessels with the six-coat system, adding varying amounts of elemental iodine (supported in glass tubes in the center of the vessel), and heating the sealed vessel at 100°C. Vapor samples were taken from the vessel and analyzed by gas chromatography with dual detection units (electron capture and flame ionization).

Results of tests at three iodine loadings are given in Table 3.3. The amount of methyl iodide detection in the vapor at the earlier sampling times was essentially independent of the amount of iodine introduced. The methyl iodide vapor concentration decreased with time and was barely detectable after 51 hr of exposure. This may have been due to decomposition of the iodide by the elevated temperature or its sorption by the paint coating. A decrease in concentration of ethyl iodide with time was

Table 3.2. Retention of Methyl Iodide on Characterization Packs with Silver Honeycomb

Relative humidity: 90%

Pack No.	Retention (%)		
	Honeycomb	High-Efficiency Filters	Charcoal
V-MP1	0.92	0.04	99.04
V-MP2	1.34	0.05	98.62

Table 3.3. Formation of Alkyl Halides by Reaction of I_2 with Amercoat 66 Paint System

Conditions		Reaction Products (mg)	
I_2 (mg)	Exposure Time (hr)	CH_3I	C_2H_5I
28.4	26	4.2×10^{-5}	2.3×10^{-4}
28.4	51	1.1×10^{-5}	1.9×10^{-4}
28.4	120	(a)	0.3×10^{-4}
10.7	4	4.2×10^{-5}	(a)
10.7	51	(a)	(a)
10.7	120	(a)	(a)
1.4	4	6.2×10^{-5}	(a)
1.4	51	(a)	(a)
1.4	120	(a)	(a)

[a]Below detection limit (about 2×10^{-6} mg per 1200 ml) for this experiment.

also noted. Previously the existence of higher iodides in the homologous series was reported in metal systems painted with either the Amercoat primer, the epoxy, or the seal gloss. However, in the six-coat paint system the organic materials released to the test atmosphere produced so complex a chromatogram that only methyl and ethyl iodide were clearly distinguishable. Undoubtedly the high-molecular-weight iodides, such as propyl and butyl, were present but indistinguishable in the gaseous mixture.

4. DIVISION OF PRODUCTION

4.1 IGNITION OF CHARCOAL ADSORBERS IN REACTOR ACCIDENTS (AEC Activity 02 30 02 90)

R. E. Adams   R. P. Shields

The effects of fission products on the ignition temperature of charcoal adsorbers are being measured. All component parts for the third in-pile charcoal ignition experiment described in the previous report were assembled and tested and are now being incorporated into the reactor unit. This experimental unit will have two charcoal adsorbers (containing MSA 85851, an iodized charcoal): one in which heat will be applied to ignite the charcoal while fission products are passing into it, as in previous experiments, and the second with vacuum-insulated walls to prevent heat loss. The insulated adsorber is included for the purpose of measuring the heat generation in charcoal by adsorbed fission products and, following the experiment, to produce information regarding the fission-product distribution throughout the charcoal adsorber. Laboratory equipment for measuring the thermal conductivity of the current activated charcoals is nearing completion.

The results of a study that relates the ignition temperature to several parameters were published.[1]

Charcoal ignition temperature is descriptively defined as the temperature above which oxidation is self-sustaining. Ignition temperature can be quite precisely measured and reproduced. In our study, ignition temperature is defined in terms of a simple theoretical model. The result is a working equation that relates ignition temperatures to several parameters. These include the charcoal activity (the effect of promoters, inhibitors, and surface area), oxygen activity (air velocity, oxygen concentration), the stoichiometry of the chemical reaction, and the activation energy of the rate-determining step of the mechanism. Effects of surface area, ash content, and air velocity were correlated with the working equation as the basis.

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[1]R. J. Davis, The Significance of Charcoal Ignition Temperatures, USAEC Report ORNL-4129, Oak Ridge National Laboratory, July 1967.

5. GAS-COOLED REACTOR PROGRAM — CHEMICAL REACTIONS

5.1 STEAM-CARBON REACTION AND FISSION-PRODUCT RELEASE AND TRANSPORT STUDIES

(AEC Activity 04 60 10 01 1)

J. E. Baker G. M. Hebert
C. M. Blood L. G. Overholser

The steam-carbon reaction and the fission-product release and transport that would occur due to accidental introduction of steam into the core regions of HTGR systems are being examined concurrently in small-scale laboratory facilities. Two high-temperature furnaces with auxiliary equipment are being employed in these studies — one for the steam-graphite reaction and the other for the reaction of pyrolytic-carbon-coated fuel particles with water vapor.

Oxidation studies of fuel segments containing bonded coated fuel particles encased in graphite were continued in the temperature range of 1000 to 1200°C with steam-helium mixtures having a partial pressure of approximately 150 torr. The simulated fuel segment consists of a graphite container 0.75 in. in diameter and 1 in. long that encloses a bonded bed of coated fuel particles 0.45 in. in diameter and 0.65 in. long. In previous studies,[1] fuel segments were oxidized at 1000, 1100, and 1200°C in a partial pressure of steam of approximately 20 torr. More recently a fuel segment was oxidized at 1100°C for 1 hr in a partial pressure of steam of 150 torr. This resulted in a burnoff of 5.5 wt %. Reaction of the steam occurred primarily in the graphite container, and sectioning of the fuel segment showed no visible evidence of attack of the residual binder material, filler, or the pyrolytic-carbon coatings of the fuel particles. A comparison of the reaction rates obtained at this partial pressure of steam with those found earlier[1] when using a partial pressure of steam of approximately 20 torr indicates that the apparent order of the reaction with respect to steam pressure is less than unity and probably is about 0.6 in this range of partial pressures.

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[1]J. E. Baker et al., ORNL Nuclear Safety Research and Development Program Bimonthly Report for May—June 1967, USAEC Report ORNL-TM-1913, p. 38, Oak Ridge National Laboratory, July 1967.

Hollow cylinders of EGCR moderator graphite, which had been impregnated with barium and heat treated at 1400°C in dry helium, were reacted with steam at 1200 and 1400°C in a partial pressure of steam of approximately 20 torr. As expected, the steam-graphite reaction was catalyzed by barium. Examination of the tube containing the graphite specimen and of the deposition tube proper for ^{133}Ba showed that some barium had migrated from the graphite to the holder. There was no evidence, however, that barium had migrated downstream into the deposition tube during the 1-hr exposure to water vapor at either 1200 or 1400°C.

Studies of the oxidation of pyrolytic-carbon-coated fuel particles were continued at 1100 and 1200°C with helium-water vapor mixtures containing 250 to 1000 parts per million by volume of water vapor. These studies are to be concluded shortly, and oxidation studies of bonded coated fuel particles will be started for which the same equipment and similar experimental techniques will be used.

5.2 IN-PILE STUDIES OF REACTIONS OF FUELED GRAPHITE WITH STEAM UNDER HTGR ACCIDENT CONDITIONS

(AEC Activity 04 60 10 01 1)

S. H. Freid   B. F. Roberts

Data reduction and evaluation continued on the results obtained from the first in-pile experiment with an HTGR-type fuel element supplied by General Atomic, as described in the last report.[1] This was a survey type of experiment and consisted of passing a moist (2 mole % H_2O) helium gas stream over the graphite fuel piece intermittently for 52.5 hr. The surface temperature of the graphite was 1050°C during most of the experiment.

Although a large number of fuel spheres had broken coatings or were unclad, the fission-product release was not large. Table 5.1 lists the percentages found of several isotopes.

The ThO_2 section contains the fuel piece, the ZrO_2 section surrounds the ThO_2 pieces, and the hot wall insulates the ceramics (i.e., ThO_2 and ZrO_2) from the primary containment. These components are located in the furnace area of the experimental assembly.

[1] C. E. Miller, Jr., S. H. Freid, and B. F. Roberts, ORNL Nuclear Safety Research and Development Program Bimonthly Report for May–June 1967, USAEC Report ORNL-TM-1913, p. 46, Oak Ridge National Laboratory, July 1967.

Table 5.1. Fission Product Retention in In-Pile Experiment with an HTGR-Type Fuel Element

	Fission Product Retention (%)			
	I^{131}	Ba^{140}	Zr^{95}	Cs^{137}
Fuel	81.9	92.0	96.6	75.8
ThO_2 sections	5.7	7.3	3.0	18.5
ZrO_2 sections	6.6	0.5	0.3	4.7
Hot wall	1.5	0.1	0.1	0.5
Released from high temperature zone	4.3	0.1	0.0	0.5

A second experimental assembly of this series was installed in the ORR reactor to build up an inventory of fission products. The conditions tentatively set for the experiment are exposure to 20 mole % H_2O vapor in the helium stream for 1 hr with a surface temperature of the graphite of 1050°C.

5.3 ENGINEERING-SCALE STEAM-GRAPHITE REACTION RATE EXPERIMENT (AEC Activity 04 60 10 01 1)

T. S. Kress F. H. Neill
E. R. Taylor

An engineering-scale steam-graphite reaction rate experimental facility is being built to investigate reaction rates under reactor accident conditions and to provide a means for corroborating analytical models developed to interpret laboratory-scale reaction rate data in terms of a reactor core geometry and flow conditions.

Steam-Graphite Reaction Rate Test Facility

Initial operation of the test facility to determine its operating capabilities was begun on July 7, 1967. During this shakedown test, a rapid depressurization occurred through failure of a seal for a graphite heater bus bar. As a result of the depressurization, heater rods were broken, and the ceramic components suffered considerable damage.

The facility is now being repaired with modifications to prevent recurrence and to enhance the assembly process. Operation should start again about September 1, 1967.

Analysis of the Steam-Carbon Reaction

A final draft of the report[1] describing the computer program Steamcar was completed and submitted for publication. In the meantime, the computer program is being examined to see whether it can be used to describe the steam corrosion of graphite in two-region geometries with each region consisting of a different type of graphite, such as fuel and moderator. Since Steamcar was designed as a special-purpose program in which the geometry is continuous, considerable difficulty has been encountered in making this further application. Some reprogramming appears necessary to meet this requirement.

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[1]D. Licke and N. Tunstall, A Numerical Solution of a System of Ordinary and Non-Linear Partial Differential Equations, USAEC Report K-1724, Oak Ridge Gaseous Diffusion Plant (in press).

5.4 HTGR NUCLEAR SAFETY TESTS IN ORR POOLSIDE CAPSULE FACILITY (AEC Activity 04 60 10 01 1)

J. A. Conlin C. L. Segaser

The ORR poolside capsule experiment was designed to measure and characterize the release of nongaseous fission products from HTGR fuel during a simulated accident that results in a temperature excursion of the fuel. A "standard" fuel, in the form of UC_2 particles in a bonded matrix, was supplied by General Atomic and was used for this and other Nuclear Safety Program studies to facilitate comparison of results.

The first poolside capsule, 06-11, was installed in the ORR May 3 and operated with a fuel-graphite interface temperature of 1050°C and a corresponding central fuel temperature of 1220°C until June 1, when a leak developed in the primary containment of the capsule. It was necessary to turn off the trap and sweep-gas line heaters and to retract the capsule to a position where the power was about 34% of normal. Prior to capsule removal a modified accident simulation was made. For this experiment the capsule temperature was brought to "normal" conditions, 1050°C in the graphite-fuel interface, 32 hr before the reactor shutdown. Nine hours before the shutdown the capsule temperatures were raised to the planned accident condition, 1450°C in the graphite-fuel interface (1720°C central fuel temperature), by advancing the capsule further toward the reactor. These conditions were held until the reactor shutdown June 16, when the capsule was removed from the reactor.

The capsule has been disassembled and postirradiation analyses are under way to determine the distribution of fission products. Analyses of components other than the fuel have so far not revealed appreciable amounts of any fission products in locations outside the fuel; however, small amounts of ^{89}Sr, ^{95}Zr, ^{103}Ru, ^{131}I, ^{132}Te, ^{137}Cs, ^{140}Ba, ^{141}Ce, and ^{144}Ce have been found. Not all capsule components have been examined, and analyses of the results are still in progress.

6. PRESSURE VESSEL AND PIPING TECHNOLOGY

6.1 HEAVY-SECTION STEEL TECHNOLOGY PROGRAM (AEC Activity 04 60 80 03 1)

F. J. Witt

A program to evaluate the effects of flaws, variation of properties, stress raisers, and residual stresses on the strength and structural reliability of present and contemplated light-water reactor pressure vessels was initiated during 1967 at Oak Ridge National Laboratory by the United States Atomic Energy Commission. The program is designated the Heavy-Section Steel Technology Program (HSST) and is to be carried out in very close cooperation with the materials, fabrication, and design segments of the nuclear power industry.

The program arose as a result of efforts by the USAEC to resolve questions posed by the Advisory Committee on Reactor Safeguards concerning various safety aspects of light-water reactor pressure vessels currently designed or contemplated for future construction. The program undertaken has evolved from proposals and recommendations from several research facilities and professional organizations, the largest contributor being the Pressure Vessel Research Committee of the Welding Research Council. The effort is supplemented by several research programs carried out by the USAEC and by cooperative efforts of the nuclear power industry.

The program is mainly concerned with welded vessels fabricated from thick-section plates (possibly up to 14 in. for some applications). Although eleven tasks have been identified under the HSST program, these tasks may be grouped under the following five general categories: program administration, material properties and material integrity, fracture behavior, periodic proof testing, and simulated service tests. Significant progress has been made in several of these areas, as reported below. Results from the material testing portion of the program should be forthcoming soon.

Program Administration

Some six to eight subcontracts are to be negotiated during the year. The first semiannual progress report for the HSST program is currently being prepared. In addition, a preliminary guide was prepared for identifying and controlling the materials included in the HSST program. A report is also being prepared that documents the fabrication history and inspection of the first ASTM A 533 grade B class 1 plate to be examined.

Materials Properties and Mechanical Integrity

Material from the first of four large plates (weighing approximately 55 tons each) of ASTM A 533 grade B steel is now available to the program. Six large specimens (92 × 14 × 12 in.) were shipped to the Naval Research Laboratory where they will be tested in a 180,000 ft-lb drop-weight tear-test machine. Specimens for the material characterization studies are presently being machined for ORNL. Heat treatment and rolling reduction efforts are under way on the three remaining plates.

The technical portions of studies on size effect in the drop-weight test and on periodic proof testing were agreed upon. Contracts for these efforts are currently under way. Six proposals for elastic-plastic analyses as related to fracture mechanics were received and are currently being evaluated, as are several other proposals for material testing.

The comparison of ultrasonic indications before and after quenching and tempering of the first A 533 plate showed that in general the number and size of indications were greater after the heat treatment. These results are still being evaluated. In this connection, the portion of the plate that contained most of the ultrasonic indications is now at ORNL. The plate was sand blasted and examined from both sides. The results in general agreed with those obtained by Combustion Engineering.

Simulated-Service Tests

Design and stress analyses of the simulated-service test vessel are continuing. A typical plot of stresses, normalized to the circumferential membrane stress of the 12-in.-thick cylindrical section, is presented in Fig. 6.1 for the conceptual design having a 2-in.-thick test course. Visits were made to most of the potential simulated-service test vessel fabricators to assess the capability of the industry with respect to both cost and scheduling.

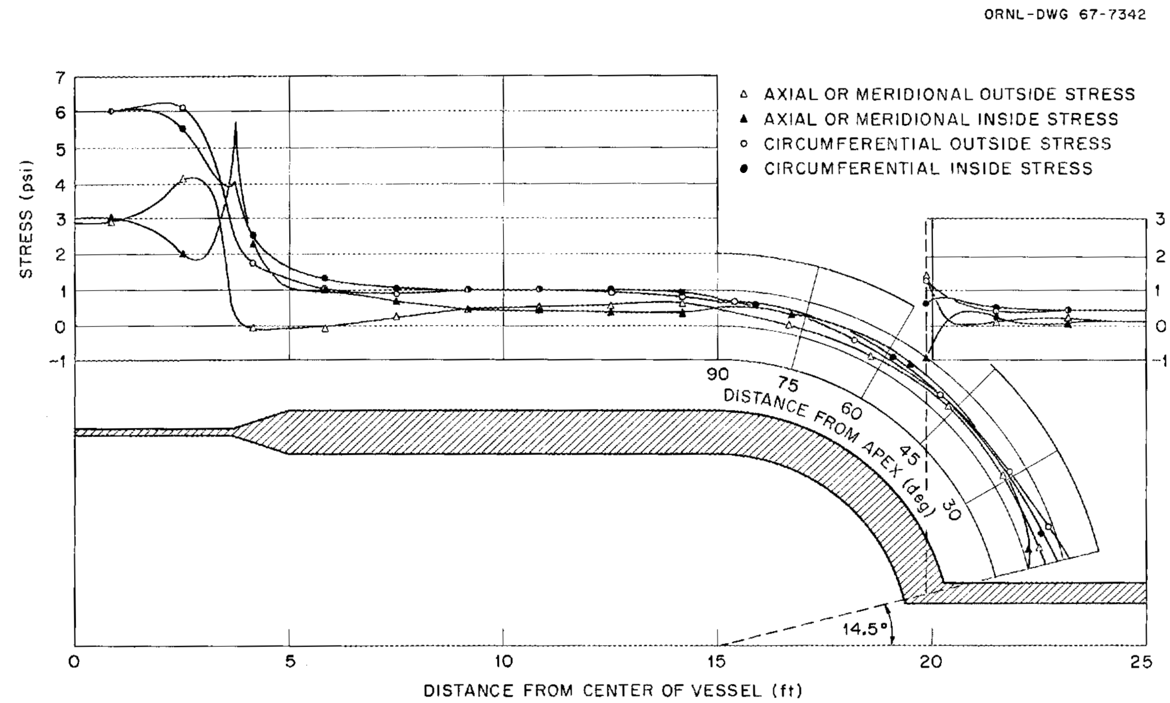

Fig. 6.1. Pressure Stresses in a Simulated-Service Test Vessel with a 2-in.-Thick Test Course Normalized to the Circumferential Membrane Stress in the 12-in.-Thick Cylindrical Section.

6.2 EXPERIMENTAL AND ANALYTICAL INVESTIGATIONS OF NOZZLES (AEC Activity 04 60 70 01 1)

B. L. Greenstreet R. C. Gwaltney

The experimental investigations of nozzles radially and nonradially attached to spherical shells and flat plates are continuing. Instrumentation and model vessels are being prepared for testing at both the University of Tennessee and Auburn University. In addition, analytical studies of nozzles attached in clusters are continuing at Applied Technology Associates.

Two spherical shell models with single nozzles attached are being examined at the University of Tennessee. The nozzle attached at 45 deg was welded in place and is ready for hand finishing at the fillet. The second model with a radially attached 7 7/8-in.-OD nozzle is being prepared for testing.

The single nozzle model for the flat-plate studies at the University of Tennessee was machined, and the other four models are in an advanced stage of fabrication. A second standard plate was machined and is ready for testing. The second loading frame for external loads is now being built.

Fabrication of the second spherical model containing a series of cluster holes is continuing at Auburn University. The strain gages that were installed on the first model with the large radial nozzles had to be replaced because of poor bonding to the metal surface. A new method of installing the gages was developed.

The analytical studies of flat plates with holes being done by Applied Technology Associates are continuing with current emphasis on third dimensional variation through the thickness in an attempt to resolve differences between present theory and experiment.

Computer programming for the cylinder-to-cylinder configuration shell problem is continuing at ORNL. Errors in the solution for the out-of-plane bending case derived by General Technology were corrected. The corrected equations are now being programmed.

6.3 DESIGN CRITERIA FOR PIPING, PUMPS, AND VALVES (AEC Activity 04 60 80 03 1)

B. L. Greenstreet S. E. Moore
T. G. Chapman

The Pressure Vessel Research Committee defined a research program for the development of stress indices for piping and for pump and valve bodies that is sponsored in part by the USAEC at ORNL. The PVRC program is intended to provide vital information to code-writing bodies and consists of 17 individual tests, several of which have been initiated, as previously reported.[1]

Literature Survey

We have initiated a literature survey on the structural design basis of piping components. The purpose of this survey is to provide a summary of research work on structural design of piping systems for background information. A report will be prepared that briefly describes what is available in the literature and provides an extensive bibliography (probably 400 to 600 references). This report is not intended to give detailed design information, but it will provide a compilation of existing design information presently in the public domain. The report will also prove useful in planning future work to fill in pertinent design information. An outline and a tentative table of contents of the report were completed. Drafts of the sections titled "Introduction and Scope" and "Factors Involved in the Design of Piping Components" were written.

Dimensional Survey of Commercial Components

The previously initiated dimensional survey on purchased commercial piping components was continued. The survey is being broadened to include manufacturers' data. Specific information on concentric reducers was requested from a list of major manufacturers. Six manufacturers have responded with data, and a seventh has indicated he will reply later. This

[1]B. L. Greenstreet, S. E. Moore, and T. G. Chapman, ORNL Nuclear Safety Research and Development Program Bimonthly Report for May–June 1967, USAEC Report ORNL-TM-1913, p. 58, Oak Ridge National Laboratory, July 1967.

information is vital for establishing stress indices and flexibility factors, since applicable standards permit a wide latitude for piping components. For example, concentric reducers can have any shape between the extremes shown in Fig. 6.2 and still fall within the specifications of ASA B16.9. A similar condition exists for all components fabricated to ASA B16.9 specifications.

Development of Stress Indices and Flexibility Factors

Parametric studies were expanded for establishing stress indices and flexibility factors as a function of applied moments, internal pressure, and dimensional variations for the ASA B31.7 code and the AEC-RDT standards. The studies are based on theoretical methods in the literature for elbows, concentric reducers, local loads (lugs, anchors, supports, etc.) on straight pipes, flanged joints, and tapered wall-transition joints. Computer codes are being prepared to facilitate computations for defining the indices and flexibility factors. Two computer codes are operable and three are partially completed. Some preliminary results have been obtained. Additional analytical methods will be programmed as the information becomes available. The parameter studies will also reflect the dimensional survey findings as they become available.

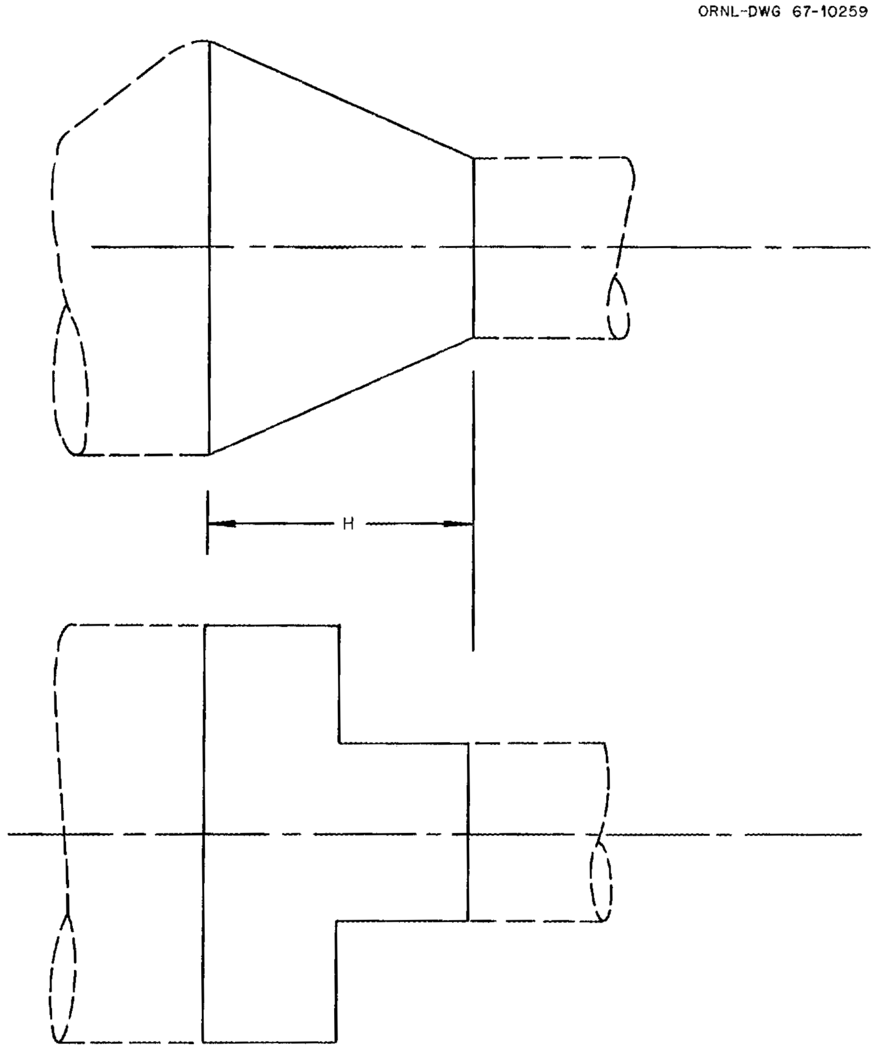

Fig. 6.2. Examples of Extreme Contours of Concentric Reducers Acceptable Under ASA B16.9 Requirements.

6.4 PRESSURE VESSEL TECHNOLOGY REPORT
(AEC Activity 04 60 80 03 1)

G. D. Whitman

The final draft of the report "Technology of Steel Pressure Vessels for Water-Cooled Power Reactors," ORNL-NSIC-21, is being prepared following incorporation of many of the comments received. It is expected that the report will be issued during the next report period.

7. ANALYSES AND EVALUATION – CODES AND STANDARDS

7.1 HTGR SAFETY PROGRAM OFFICE
(AEC Activity 04 60 70 01 1)

S. I. Kaplan M. D. Silverman
E. A. Nephew W. C. Ulrich

The HTGR Safety Program Office continued its evaluation of current research and future program needs in HTGR physics, coolant chemistry, and fission-product behavior. An analysis of system and component performance begun during the previous report period is continuing.

High-Temperature Physics

The subcontract with Pacific Northwest Laboratory (PNL) involving the use of the High-Temperature Lattice Test Reactor (HTLTR) as a facility for transient tests of HTGR fuel elements was signed, and a preliminary survey was started by PNL to select appropriate parameters for the study. The extent of preparation and the general approach to be followed were reviewed at a meeting between the HTLTR staff and a Program Office representative at the HTLTR site on August 2.

We are continuing to study the possible usefulness of the UHTREX reactor for examining transient reactivity feedback characteristics in HTGR fuels. Our intent is to calculate the peak power following a step insertion of reactivity for several illustrative UHTREX fuel arrangements involving both segregated and homogenized fissile and fertile materials. If a significant variation in peak power were attainable by varying the fuel configuration within practical limits, UHTREX could, in principle, be used to study the effects of transient heat flow within HTGR fuel compacts on core reactivity.

Coolant Chemistry and Fission-Product Transport

A summary report on fission-product transport and coolant chemistry is being prepared for USAEC-DRDT. Toward this end, a Program Office representative conferred with research staff members of PNL and General Atomic (GA). Discussions at PNL concerned the relevance of their work in this field to the evaluation of HTGR accident situations and their plans for future work. Topics reviewed at GA concerned experimental means for simulating potential HTGR safety problems, including carbon deposition and steam-graphite reactions and for predicting fission-product distribution under normal and abnormal operating conditions.

Allied Activities

The study of HTGR systems and components initiated during the last report period is continuing. We are also maintaining our coverage of ACRS discussion meetings on the Fort St. Vrain HTGR.

7.2 NUCLEAR SAFETY INFORMATION CENTER
(AEC Activity 04 60 70 01 1)

J. R. Buchanan Wm. B. Cottrell

The Nuclear Safety Information Center was established by the U.S. Atomic Energy Commission, Division of Reactor Development and Technology to collect, evaluate, and disseminate nuclear safety information for the benefit of the nuclear community. In order to do this we found it necessary to computerize reference sources, and we are now in the process of installing a telecommunication station to expedite our various uses of this information file. While many members of the nuclear community contact the center for assistance ranging from the identification of a single report to special literature searches, as well as consultation and special studies, a major fraction of our effort is in the preparation of state-of-the-art reports and our quarterly bibliography.

Telecommunications. The equipment for NSIC's computer telecommunications station was received. The remote facility consists of an IBM-2740 typewriter-printer and two IBM-2260 display stations. Installation of the 2740 was completed and programs were written and are being tested on an abbreviated NSIC file. These programs are in the final stages of debugging. Installation of the 2260 display stations awaits delivery of the coaxial cable needed to tie the stations to their control unit. Programming checkout of the stations will be initiated upon completion of the equipment installation.

Reports. The Center's tenth quarterly indexed bibliography was issued in August as ORNL-NSIC-36.[1] It contained 1189 references sorted into 19 subject categories. Also issued in August as ORNL-NSIC-35[2] was the latest version of the Center's keyword thesaurus. It contains over 1500 keywords

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[1] NSIC Staff, Indexed Bibliography of Current Nuclear Safety Literature-10, USAEC Report ORNL-NSIC-36, Oak Ridge National Laboratory, August 1967.

[2] NSIC Keyword Thesaurus, USAEC Report ORNL-NSIC-35, Oak Ridge National Laboratory, August 1967.

developed for use in indexing and retrieving the information cataloged by NSIC and reflects the experience gained by 18 specialists in indexing over 12,000 documents.

The third edition of the "Compilation of National and International Nuclear Standards (Excluding U.S. Activities)" was completed and submitted to the printer. It will be issued in September.

First drafts of two state-of-the-art studies, "Fission Product Release and Transport in Liquid Metal Fast Reactors" and "Sources of Tritium and Its Behavior Upon Release to the Environment," were completed and are being circulated for initial review and comments. Meetings to consolidate comments and make plans for revisions of the drafts will be held in September.

Other Information Activities. During the reporting period, NSIC received and answered 86 information requests. In addition, there were 18 visits to the Center for staff consultations. The AEC's "General Design Criteria for Nuclear Power Plant Construction Permits" were reviewed by the NSIC staff, and the comments and suggestions were submitted to the Commission. During the past two months, 43 persons were added to our SDI (bimonthly Selective Dissemination of Information service). These persons brought the total number of participants to 824.

7.3 DISCUSSION PAPERS ON VARIOUS ASPECTS OF WATER-COOLED REACTOR SAFETY

(AEC Activity 04 60 80 03 1)

J. W. Michel

The Laboratory is preparing a number of discussion papers on various aspects of water-cooled reactor safety. The papers, directed to technical management, are intended to summarize both current practice and current technology and to identify the need for additional research in order to understand better or otherwise further enhance the safety of water-cooled power reactors. The subjects of these papers were selected from among the most urgent safety questions of the day and include in-core cooling, metal-water reactions, earthquakes and plant design, shock and missile considerations, air cleanup systems, containment system testing, and reactor plant instrumentation systems.

Extensive comments were received on all the discussion papers following their distribution last spring, and the papers are being revised accordingly. All the external review committees of experts, which were assembled for each paper, completed their reviews, except the committee on reactor protection instrumentation systems, which is scheduled to meet in September. Their advice on the extent and nature of the revisions required to the draft sent out for comment was obtained. With some few exceptions, the response from those who were asked to comment on the report drafts has been poor. Additional efforts to obtain more input of this form from the nuclear industry are being made. In the meantime, revisions to the papers are being made in preparation toward the completion of the final reports.

The comments received to date include several that would significantly extend the scope of this work, others that question the propriety of comparing the designs and analyses of different manufacturers, as well as numerous others on relevant technical details. At this time we have received no written comments from the Commission, but these are expected in September. Review meetings with the Commission will follow on each paper. Unless the Commission authorizes a significant extension of the scope of any paper, it is expected that the final reports will be published in about four months after these final review meetings.

7.4 FUEL TRANSPORT SAFETY STUDIES
(AEC Activity 04 60 80 03 1)

L. B. Shappert

A code is being developed to set forth suitable engineering practice standards for the design, fabrication, testing, inspection, and maintenance of irradiated-fuel shipping casks. The code is to provide design information on the structural shielding, heat transfer, criticality, materials, fabrication, and inspection methods. In addition to the code, a backup document is being prepared that will contain detailed justification, derivations, and judgments on the code materials and will indicate the degree of safety and degree of conservatism specified by the code. As a part of the program, tests are conducted, where necessary, to resolve important issues. Recently, members of the ORNL program participated in an AEC review of current shipping cask designs developed by various manufacturers.

Preparation of the shielding chapter of the criteria brought the number of completed chapters to four. These are the chapters on materials, fabrication, criticality, and shielding. The chapter on heat transfer, which was submitted in June by Battelle Memorial Institute, is presently being rewritten to conform to the style of the criteria.

Heat Transfer Criteria

Further efforts on thermal model simplification were deferred pending implementation of the THT-D computer program on Battelle's Columbus Laboratories CDC 6400 Computer. The THT-D computer program is currently going through a "debugging" period, and we expect it to be ready for use soon.

Supplemental detailed temperature mappings for a 12 by 12 fuel element array will also be generated when THT-D becomes fully operable.

At present, the fire analysis outlined in the Cask Acceptance Criteria Manual does not account for voids left in the ends of the cask for the purpose of accommodating lead expansion. Leaving voids at these places is common practice among some cask fabricators. Since the void spaces have an insulating effect during a shipping fire, the edges of the cask will not melt as fast as if no voids were present.

In order to account for these voids in the fire analysis procedure, two analytical methods are being investigated. The first method utilizes the Newman equations (explained in Volume I of the Acceptance Criteria Manual). In this method the steel at the corner of the cask is treated essentially as if no void were present, and the temperature distribution at the corner is computed. From the calculated temperature distribution at the corner and radiation heat transfer equations, it is possible to account for heat transmission across the void. When the solid region is reached across the void, the Newman equations are used again.

A somewhat more sophisticated method for computing the perturbation of heat transfer due to voids is to use three-dimensional finite difference equations. This second method does not involve the degree of approximation inherent in the first method; however, it would be somewhat more formidable for a hand calculation. Since a Fire Analysis Computer Program now exists, the second method may be used in the program, and the simpler method may be recommended for investigators not having access to a computer.

Most manufacturers of casks provide expansion voids that are supposed to be filled in the event the cask is involved in a fire accident. This design feature is intended to prevent thermal stresses from building up in the cask (due to lead expansion upon melting) that would cause it to rupture. There is some controversy concerning the ability of lead, once it is melted, to flow into the voids provided. The studies being carried out are of an analytical nature and are an attempt to resolve the controversy surrounding the worth of voids as presently designed in lead shields.

Review of Current Designs

Elk River Shipping Cask. The review of the structural analysis of the Elk River shipping cask was completed. It was concluded that no guarantees could be placed on the retention of the cask closure seal in a 30-ft free fall. A secondary problem is that upon solidification after a fire, if the shell has been strained plastically and maintained its integrity, a potential exists for the creation of voids. It was recommended that two silicon O-rings be used with this cask, that protection boxes be designed for the thermocouple wells, and that a general inspection be made to determine whether any areas of the cask require repair.

Hallam Shipping Cask. Over the past several months a question has arisen as to the importance of obtaining a good bond between the lead shielding and the outer steel shell of the Hallam shipping cask designed by Atomics International. The safety analysis was based on the fact that a bond exists. The company that is pouring the lead will not guarantee any bond. It seemed imperative, therefore, to build a model cask and perform some tests to determine whether the movement of lead in a 30-ft impact was severe enough to actually require a bond.

The model cask, designed at ORNL, built in Knoxville, and tested at the University of Tennessee was about 32 in. long and 5 in. OD; the scale was approximately 7.5 to 1. The cask did not impact in the truly vertical position but was canted at an angle of several degrees. This created a slight bulge on one side of the cask about 2 in. above the point of impact. Other than that, deformation seemed to be relatively uniform and the inside diameter changed only slightly. The lead settled approximately 0.7 in. This corresponds quite well with the 0.40-in. settling obtained by H. G. Clarke at Franklin Institute when he tested an 18-in.-long 3-in.-OD model cask. As a result, there seems little doubt that the Hallam cask will have to be checked ultrasonically to determine the amount of bond that exists after the cask has been poured.

We therefore decided to examine the effect of lead-steel bonding on impact results, and Franklin Institute built a second model cask identical to their first one. The inner surface of the outer pipe was tinned on alternate quadrants of the cylinder to insure reasonable lead wettability over 50% of the total inner surface area. Visual inspection indicated that a good bond was obtained in the quadrants that were tinned, and no bond was obtained in the quadrants where no tinning was provided. Lead movement was considerably reduced over that in the completely unbonded cask when the cask was subjected to a 30-ft drop test. The lead settled an average of only 0.06 in. in the partially bonded cask, with settling as small as 0.005 in. in one of the bonded quadrants.

The fabrication of the second model cask at the University of Tennessee was completed. This cask incorporates the "Knapp Mills" closure design on either end, which is characterized by the loading of all bolts in compression and massive flanges and plugs. Testing of this cask under 30-ft impact conditions will begin shortly.

TRU 10-Ton $^{252}Cf/^{254}Cf$ Carrier. Limonite-concrete bricks of the type to be used in the TRU cask were received, and the type and means of testing are now being formulated. It is proposed to perform both 40-in. puncture tests and 30-ft drop tests of these bricks to determine the extent of damage of unreinforced concrete. Detail design of this carrier was completed and is now under review. A design report was started that will demonstrate the compliance of this cask with AEC Manual, Chapter 0529.

The 2600-lb concrete cask for shipping curium isotopes was examined to determine compliance with AEC Manual, Chapter 0529. Results indicate that the cask will meet the regulations, and we expect to obtain approval for shipping a number of transuranium isotopes.

Navy Shipping Casks

We are examining the problems of the spent-fuel shipping cask for the PM-1 and PM-3A reactors and recommending solutions. Accordingly, the Navy sent us a fresh fuel shipping container for testing. The container is approximately 2 1/2 ft in diameter and 4 ft tall and will be tested by dropping it 30 ft onto a corner.

Iodine Release in Shipping Accidents

The release of fission-produced iodine from UO_2 to the free volume of the surrounding sheath is a function of the initial density, the burnup level, and the operating temperature of the UO_2. In low-density fuels of 80 to 92% theoretical density (100% UO_2 density = 10.96 g/cm^3) the observed release rate of iodine (temperature below 1600°C) is comparable with that of xenon; however, for high-density fuels (greater than 92% theoretical) the release of iodine exceeds that of xenon. As the present time the usual reactor density for UO_2 fuel is in the range of 93 to 94% theoretical, and this density range will be the reference for the following discussion of the effect of temperature and burnup on iodine release.

Data for the analysis of the temperature and burnup effect on iodine release were obtained by Parker[1] by heating reactor-grade UO_2 in very pure helium for a period of 5.5 hr. The data indicated that the release of iodine increased with increasing burnup and increasing temperature. Appropriate curves for the estimation of iodine release are given in Fig. 7.1, where the percentage of iodine released is plotted against the reciprocal of the absolute temperature. The solid lines (tracer, 1000, and 4000 Mwd/T) represent actual data obtained for those exposure levels; however, the curves for 5,000 and 10,000 Mwd/T are predicted from previous data and have not been substantiated by experiment. The curve for the 20,000-Mwd/T burnup is from data on the burnup of PuO_2-UO_2 fuel collected by Nuclear Materials and Equipment Corporation (Quarterly Progress Report, December 1966). The curves in Fig. 7.1 may be used for the estimation of iodine gas release from UO_2 fuels during shipping accidents, and the curves may be extended for lower temperatures. For temperatures greater than 2200°C the release of iodine approaches 100%.

International Symposium

A letter was submitted to AEC-ORO from Union Carbide Corporation offering to act as host for a Second International Symposium on the Transportation and Handling of Radioactive Materials. Some groundwork was laid in anticipation of the AEC's approval. We expect to attract approximately 400 people to the meeting, and, consequently, we have arranged to obtain the Civic Auditorium at Gatlinburg for October 14–18, 1968. This should give us sufficient time to organize an excellent program covering all aspects of the transportation business.

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[1]G. W. Parker et al., Out of Pile Studies of Fission Product Release from Overheated Reactor Fuels at ORNL 1955–1965, USAEC Report ORNL-3981, Oak Ridge National Laboratory, July 1967.

ORNL DWG 67-6946

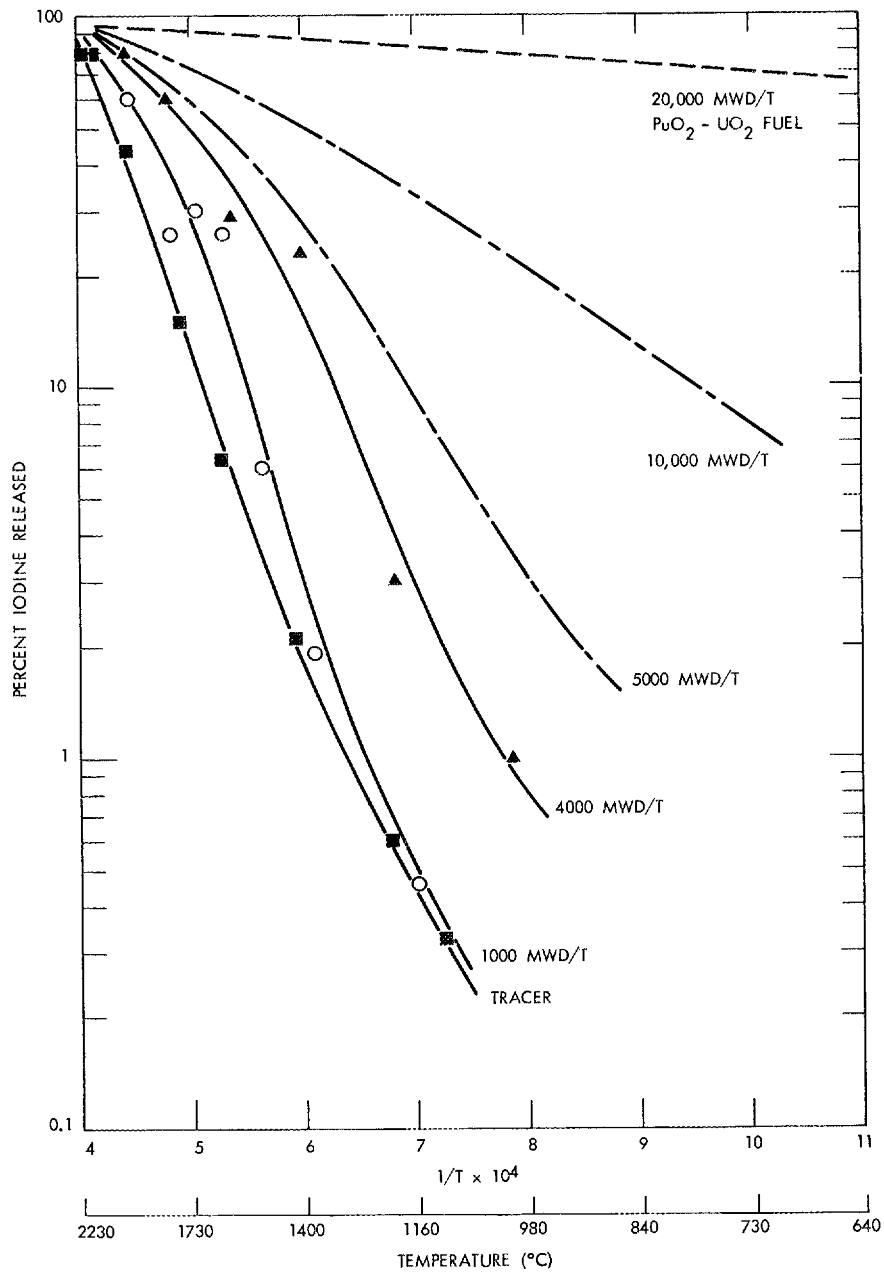

Fig. 7.1. Percent Iodine Release from UO_2 by Diffusion in 5.5 hr as a Function of Reactor Operating Temperature.

7.5 COMPUTER HANDLING OF REACTOR DATA FOR SAFETY (CHORDS) (AEC Activity 04 60 70 01 1)

W. E. Browning, Jr. D. W. Cardwell

A computerized system is being developed for storage and retrieval of design, construction, and operating safety characteristics of nuclear reactor facilities, particularly those of greatest current interest to the AEC Division of Reactor Licensing.

Structuring of information for computer input is continuing for areas of containment, safety controls, design basis accident, and containment cooling system on the first seven water-cooled power reactors. Some preliminary assessment is being made of the degree of availability of pertinent data from safety analysis reports (SAR's). Considerable effort is being devoted to reconciling the opposing requirements growing out of the need for specific information on each of a large variety of reactor systems on the one hand and the need for simple, efficient retrieval of information on the other. Several approaches are being employed toward this end. The first is to insure that only information that is the most important and is likely to be used in safety evaluations is included. This is being accomplished by the application of engineering judgment in original formulation and by internal and AEC review of the information.

Another approach takes advantage of the selectivity that is possible during computer retrieval. Information is selected according to criteria of subject matter or content that reflect the needs of the user. For example, a user concerned with a certain component of a nuclear plant will see data on only that component. As another example, a user evaluating a new reactor may be presented with only that part of the data differing from that of a similar reactor previously evaluated. Data drawn from SAR's are entered into the computer memory with designated ranges of significance appropriate to individual categories. Such designation, and the accompanying computer program, offer a rapid comparison option to potential users of the system. The program for accomplishing such comparative searches has been checked out and is now operational. This option makes it possible to scan large volumes of CHORDS information and to limit initial readout to items where significant differences occur either between data from several reactors or between data representing different conditions within a single reactor.

Certain data drawn from SAR's must be listed in alphameric or alphanumeric (as opposed to numerical) expressions. Considerable effort is being concentrated on the most efficient forms that can be used for such expressions. Progress has been made in deriving forms that reduce volume and standardize terms to encourage rapid comparison between individual characteristics of different reactors. As a result of selectivity during retrieval, the user need not be aware of the mass of data outside his needs.

Other approaches involve various techniques for refining and condensing data prior to entry into the computer file. This effort is expected to result in a system that will, as simply and efficiently as practical, retrieve the data needed by each of a variety of specialized users.

Coding and punched card entry into a central computer has been started for characteristics and data in the area of reactor containment. Programming language for initial queries is under development, and some very preliminary trials have been made to determine techniques that may be most efficiently employed for comparative searches of the data bank. It is expected that this development of query techniques will be a gradually evolving process.

A study was initiated to devise methods for collection of equipment performance data from operating nuclear power plants for computer storage directed toward reliability analysis. Discussions held on this subject with representatives of various AEC divisions show a high degree of interest in the potentialities of such a program as a means of significantly benefiting many areas of reactor safety.

7.6 RDT REACTOR STANDARDS
(AEC Activity 04 60 80 03 1)

R. A. Schmidt

The RDT Standards Program was established to provide an extensive engineering foundation for AEC reactor programs to assure that reactor designers, material suppliers, fabricators, builders, and operators will be apprised of current proven technology for the design and construction of safe, reliable reactors. A large part of the effort will be directed toward the preparation of design methodology guides for components, including vessels, piping systems, and engineered safeguards. The Standards Program will draw on both favorable and unfavorable experience and successful developments emanating from programs for reactors, systems, components, and materials within the AEC and elsewhere. Immediate efforts are being directed toward assisting AEC, its laboratories, and its contractors through cooperative efforts as requests and current needs arise. Long-range plans include preparation, distribution, and continuous updating of standards to be used for pending and future reactors. ORNL will coordinate the incorporation of standards that are being prepared and distributed at several sources under AEC auspices into the RDT Standards Program.

Program Organization

Selection and assignment of ORNL personnel to staff the Standards Group were accelerated in order to proceed with the program as it was presented in proposals to the AEC and stated in a letter from DRDT. Arrangements are being made to obtain input from other AEC laboratories, contractors, and offices for the preparation of standards where parallel activities are being carried on at those installations. A survey was also initiated, in cooperation with the NSIC and AEC, in which the RDT Standards Group will review reports from current activities where information relative to reactor standards may be obtained.

Preliminary discussions were initiated with several engineering consultant organizations that are experienced in specific phases of reactor engineering. Contracts will be arranged to obtain assistance where the needs of the program require. United Nuclear Corporation of Elmsford, New York, is presently conducting a survey and preparing a list of standards and classification of components required for primary coolant systems for PWR's and BWR's and is also working on evaluation and recommendations for updating an existing design manual.

Standards Preparation. Several existing preliminary standards were examined to serve as possible bases for an RDT reactor protection instrument system standard, and recommendations were transmitted to DRDT. None of the standards reviewed are considered adequate in their present form. Future work will further consider the existing standards as they evolve, or a new standard will be prepared if necessary.

Preparation of drafts for tentative standards based on specifications provided by AEC continued during the report period. A total of 69 documents has been submitted to DRDT for review. Several standards were approved by DRDT for distribution as "tentative," two of which were issued to AEC reactor program participants.

Special Studies. A preliminary review was made of malfunctions caused by vibration-induced tube damage in heat exchangers in AEC-sponsored reactors. Much of the information was provided by key personnel associated with the various reactor installations. A brief summary and recommendations based on the limited study were transmitted to DRDT. Investigations of this problem are continuing for the assistance of current reactor programs and for incorporation in heat exchanger standards, which will be prepared at a later data. At the request of the AEC, ORNL reviewed the AEC-ACRS document entitled "Inspection and Testing Requirements for the Reactor Coolant System and Certain Associated Systems of Nuclear Power Reactors." A commentary was prepared and submitted to DRDT to assist in the evaluation of the proposed requirements for verification of the integrity of a reactor throughout its operating life through scheduled nondestructive inspections.

7.7 SAFETY STUDIES FOR MOLTEN-SALT BREEDER REACTORS (AEC Activity 04 60 80 03 1)

R. B. Briggs E. S. Bettis
A. M. Perry J. P. Nichols
H. F. McDuffie

Studies are under way to obtain a basis for evaluating the safety of molten-salt breeder reactors. Successful operation of the Molten-Salt Reactor Experiment and encouraging results from evaluations of designs for large molten-salt power breeder reactors have led to inclusion of a program for their development in the "high-gain" breeder section of the AEC's Civilian Power Reactor Program. Molten-salt reactors are unique in that the fuel is a liquid at the high operating temperature and is pumped through the reactor core and through cooling and processing circuits external to the reactor vessel. Their safety characteristics may be more favorable or less favorable than those of gas-, water-, and liquid-metal-cooled reactors but, in general, are considerably different. The analytical and experimental work of these safety studies is, therefore, essential to further development of this type of power reactor.

The safety studies include design, analytical, and experimental work and involve studies of reactor dynamics and stability, evaluation of requirements for emergency shutdown and cooling systems, calculation of the effects of failure of equipment in the reactor primary and secondary circuits (including effects of mixing of intermediate coolant with fuel and blanket salts), and studies of fission-product disposition. Analyses of the fission-product inventories in typical plants and the effect that the proposed rapid fuel-processing rate will have on safety considerations will be studied. Small-scale experimental work on the release of fission products from surface deposits and from salt under accident conditions is planned.

This is the first bimonthly report on this program. Since the program is new this year, the early part of the year will be spent in general analysis of the safety of molten-salt breeder reactors. These studies will be followed by more detailed analysis of specific problems and by experimental work.

Reactor Dynamics

The use of a liquid fuel that is circulated through the heat exchanger system leads to safety considerations in an MSBR plant that differ considerably from those in the more conventional water reactor plants. The fuel is already at high enough temperature to be molten, but the pressure is low, fission-product burdens are low, and no violent chemical reactions can occur on mixing of the fluids with each other or with the environment.

A safety analysis will begin for a typical MSBR plant in order to define criteria for design of plants to meet AEC standards of safety. At the beginning of the analysis, the safety criteria will be established. Then, current designs of MSBR plants will be studied and accidents that have safety implications will be described. The type and degree of containment required and other means that must be provided to prevent these accidents from importantly affecting the health and safety of the public will be specified. Studies will then be started to examine how the plants can be designed to satisfy the safety criteria and how the criteria affect the cost and operability of the plants.

In analyzing the safety of a nuclear power plant, one of the major areas of concern is the dynamic response to reactivity fluctuations, as well as nuclear transients. Nuclear transients can be introduced by various means, one of major concern being the flow of fuel into the blanket region, which could conceivably occur under some accident conditions. Likewise the conditions under which small fluctuations in reactivity could cause power oscillations to build up to harmful magnitudes are of interest. Molten-salt reactors have new features whose effects on dynamics cannot be predicted quantitatively on the basis of past experience. For instance, the system will use circulating ^{233}U fuel. The fuel will have a prompt negative temperature coefficient, but the small delay fraction of ^{233}U will be reduced to an even smaller effective value by circulation. Also the system is heterogeneous, having circulating fuel and blanket, and consequently has built into it a variety of time delays, including heat transfer from graphite to fuel, fuel transport in core, blanket transport in core, and others.

At first a general study will be made of the manner in which these factors affect the stability of a simplified plant to establish ranges of interest. Then a preliminary linearized analysis will be made of current designs to obtain an indication of their stability characteristics.

Emergency Cooling Requirements

Use of liquid circulating fuel makes the emergency shutdown and cooling requirements for a molten-salt reactor considerably different from those of reactors with solid, stationary fuels. The fuel can be drained from the reactor to stop the nuclear reaction. Special means for removal of decay heat can be provided in the drain tank system. Fission products are removed from the fuel continuously on short cycles, so the decay heating is reduced. A study will be made to establish emergency shutdown and cooling requirements and to evaluate current designs.

This year we will make a detailed study of the distribution of decay heat in a molten-salt reactor plant during operation and as a function of time after shutdown. The decay heat is produced by fission products in the fuel, in the off-gas system, in the fuel-processing plant, and in deposits on the inner wall of equipment and piping and on the graphite in the core. By making use of the best chemical data and information from the MSRE and from design studies, we will calculate the distribution of fission products in the reactor and processing plant as a function of time during the 30-year life of a reference reactor. These fission-product distributions will then be converted into heat-production distributions during power operation and as a function of time after shutdown. Based on these numbers, we will specify the amount and distribution of emergency cooling that must be provided for a large power station of reference design.

Coolant Mixing

Designs for molten-salt breeder reactors generally have an intermediate coolant for transferring heat from fuel and blanket salts through an intermediate circuit to water in a supercritical steam generator. The intermediate coolant is kept at higher pressure than fuel and blanket salts so that coolant will leak into the reactor circuits if one or more heat exchanger tubes fail. Mixing with the coolant will produce changes in the properties of fuel and blanket salts and could affect the safety of the reactor.

This year we will examine reference designs of components for molten-salt breeder reactors and estimate the rates of transfer of coolant into fuel and blanket circuits for failure of one and more tubes in the intermediate heat exchangers. The resulting changes and rates of change in composition of fuel and blanket streams will be calculated and used to estimate the nuclear behavior of the reactor. Laboratory studies will be started to determine the effects of the mixing on the liquidus points of fuel and blanket salts and on the behavior of the uranium in the fuel salt.

Fission-Product Studies

The seriousness of a major accident in which fuel is spilled through a break in a pipe in a molten-salt reactor depends on the burden of fission products in the salt, the fraction released from the salt, and the fraction released from deposits on surfaces in the reactor. The distribution of fission products in a reference plant will be calculated. We also need to determine the amounts that would be released in various environments.

This year a small apparatus will be designed and assembled for conducting fission-product release studies in a controlled environment. The apparatus will be tested in a semihot area with trace quantities of radioactivity and will be modified until it performs satisfactorily. It will then be prepared for installation in a hot cell.

8. DIVISION OF TECHNICAL INFORMATION

8.1 TECHNICAL PROGRESS REVIEW NUCLEAR SAFETY
(AEC Activity 07 13 02 02)

J. P. Blakely W. B. Cottrell

Nuclear Safety is a Technical Progress Review prepared and edited at ORNL under contract to the USAEC. The Summer 1967 issue, Nuclear Safety 8(4) was distributed on August 7. It was the last to be issued on the quarterly schedule.

Nuclear Safety 8(5), September—October 1967, had been in preparation for some time prior to its recent approval as the first issue to be published on the bimonthly schedule. It therefore had to be reduced in size to more nearly conform to the new average of 100 pages per issue. (The total yearly allowance remains at 600 pages, although publication frequency was increased from quarterly to bimonthly.) These changes resulted in some loss of schedule, and final copy left DTIE on August 31, with publication due around October 1. This issue will contain 116 pages.

External review of the draft of the November—December 1967 issue, Nuclear Safety 8(6), was completed, and responses to comments were received from all but two authors. Changes were submitted to DTIE by September 1 for all but three articles. This issue will contain articles by four ORNL authors and six authors from outside the Laboratory, including two from UKAEA. Publication is scheduled for about November 1.

Nuclear Safety 9(1), January—February 1968, was distributed for review in draft form on August 31 as ORNL-TM-1930. The Feature Article that had been submitted by Dr. E. Teller was temporarily withdrawn at the request of the author. The external review of ORNL-TM-1930 should be completed by October 1, and publication is scheduled for January 1, 1968.

Articles are presently being received for Nuclear Safety 9(2), February—March 1968. The draft is scheduled for external review beginning November 1, with publication by February 1.

The working outline for Nuclear Safety 9(3) was finalized, and authors were notified. A preliminary outline for 9(4) was also developed, and possible authors are presently being contacted.

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