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ORNL NUCLEAR SAFETY RESEARCH AND DEVELOPMENT PROGRAM
BIMONTHLY REPORT FOR MAY - JUNE 1967

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ABSTRACT

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The accomplishments of the research and development program being undertaken at ORNL as part of the U.S. Atomic Energy Commission's Nuclear Safety Program during the months of May and June 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 the work has been in the past, and continues to be, in general support of water power reactor technology including some in direct support of the LOFT and CSE programs, several projects are 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 the studies of pressure vessel technology. Experimental work relative to pressure vessel technology includes investigations of the attachment of nozzles to shells and the implementation of this joint AEC-PURC programs on Heavy Section Steel Technology and Nuclear Piping Pumps and Valves. Several of the projects are concerned with the development of engineering safeguards and the establishment of codes and standards for these safeguards while 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 Nuclear Safety journal in behalf of the nuclear community are also discussed.

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1.1 CHARACTERIZATION OF SIMULATED ACCIDENT AEROSOLS FOR CSE
(AEC Activity 04 60 10 011)

T. H. Row R. E. Adams G. W. Watson

A program is in progress to characterize the fission product simulants proposed for use in BNWL Nuclear Safety Containment System Experiment (CSE). A portable sampling unit was designed to be used in experiments conducted at CSE and CMF to aid in comparing the data from each facility. The purpose of the experiments is to provide confirmation that the transport behavior of the simulating aerosol is sufficiently similar to that of highly irradiated fuel.

Preliminary analysis of the first CMF-CSE type run as indicated by the portable sampling unit has been completed. A comparison of these data with those from CMF sampling equipment shows very good agreement both in concentration and characterization.

The portable sampler was transported to BNW at Hanford and used in an experiment in the SAT facility of the ADF laboratory. The samples from this experiment are being analyzed at CSE with compilation of the data to be done at ORNL.

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

G. W. Parker R. A. Lorenz

Plans are being made to reorient the current series of fission product release experiments in TREAT from the simulation of underwater transient melting accidents to the investigation of "prompt" fission product release. "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. Clad rupture results because of high cladding temperature and high internal gas pressure following loss-of-coolant. The in-pile TREAT experiments will use fission heat to simulate the decay heat (and initial heat content of UO_2 above cladding temperature) of a real loss-of-coolant accident. Other prompt release experiments are presently planned to be conducted in out-of-pile facilities.¹

Some cladding rupture is expected to occur following rapid loss-of-coolant from high performance water reactors in spite of emergency core cooling devices such as high pressure water sprays. For example, a recent design and analysis report² for a high performance water-cooled reactor showed that the cladding of some fuel rods may rupture in less than 10 sec following major loss-of-coolant and that 75 percent of the fuel rods could eventually fail under the conditions assumed for their calculation.

The prompt fission product release experiments in TREAT will use a cluster of 7 high burnup fuel rods in a flowing steam atmosphere with helium added to carry the released radioactive material into a collection

¹G. W. Parker et al., The Prompt Release of Fission Products from Highly Rated UO_2 Fuels, ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1967, USAEC Report ORNL-TM-1864, p. 16, Oak Ridge National Laboratory, May 10, 1967.

²Browns Ferry Nuclear Power Station Design Analysis Report, Docket 50-259, Tennessee Valley Authority, Vol. I, July 1966.

train of filters, a condenser, and charcoal beds. The fuel rods will be electrically preheated to about 550°F and will receive a slow transient in TREAT to heat the rods to cladding failure in less than one minute. Thermocouples on the rods and radiation monitors on the exit gas line will determine the temperature at which the cladding fails.

The fuel planned for use in the first tests was irradiated in the VBWR until December 1963. These fuel rods are 40 in. long and contain 95 percent dense uranium dioxide pellets in self-supporting Zircaloy tubes 0.426 in. O.D. The fuel received burnup in the range of 4000 to 10,000 Mwd/T with average specific powers in the range of 20 to 50 kw/kg of uranium. Long-lived ^{129}I will be analyzed and calculations made to predict the release of ^{131}I from a "green" fuel rod. ^{131}I is the most important biological hazard and its equilibrium distribution between UO_2 and void spaces is different from ^{129}I because of faster radioactive decay.

Eighty-eight of these VBWR fuel rods are currently receiving additional burnup in Dresden and may be available in 12 to 15 months for experimental use. This would extend the burnup range and could provide "green" fuel rods with ^{131}I realistically distributed between UO_2 and void spaces.

A second source of possible fuel is from a specially controlled group of Zircaloy clad fuel rods exposed in the Yankee reactor core for one core cycle (10,000 Mwd/T). These rods are too long for TREAT and would have to be shortened. This possibility is being discussed with Westinghouse as part of their planned examination of a portion of these rods. Internal gas pressure would be measured, the fuel rods cut into desired lengths, and the rods repressurized and seal welded.

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

(AEC Activity 04 60 10 01 1)

C. E. Miller, Jr. S. H. Freid B. F. Roberts

The second experiment employing the simulated containment vessel during an in-pile fuel melting experiment has been completed. This experiment consisted of melting a Zr-clad UO_2 fuel element in a moist air atmosphere. The containment vessel was maintained at approximately 103°C throughout the meltdown and subsequent sampling period. The meltdown period was curtailed after 3-1/4 min due to excessive pressure buildup in the containment vessel. However, later examination of the fuel showed that complete melting had occurred.

Five May packs and two bulk gas samples were taken between 1 hr and 10.4 hr after the meltdown. In addition, the entire experiment was disassembled and each component is now being analyzed. The bulk gas samples were taken at 4.4 and 9.3 hr after meltdown. Both samples indicated significant amounts of ethyl and methyl iodide. Butyl iodide was present only in the earlier samples. In each case the amount of ethyl iodide present was at least twice as great as the methyl iodide.

Preliminary results have been obtained for some of the May pack samples. It appears that the behavior of the iodine species are different in the hot containment vessel ($\sim 100^\circ\text{C}$) than in the cold vessel ($\sim 40^\circ\text{C}$). This difference is most noticeable in an apparent increase with aging in the amount of iodine which is deposited in the charcoal papers and beds.

1.4 CHARACTERIZATION, CONTROL, AND SIMULATION OF ACCIDENT-RELEASED FISSION PRODUCTS

(AEC Activity 04 60 10 01 1)

R. E. Adams R. D. Ackley Zell Combs F. V. Hensley

The purpose of this program is to investigate the characteristics of those gas-borne fission products likely to be released in nuclear accidents and to study their behavior under the various conditions which would be expected in accidents to water-cooled reactors. Activities during this period included study of the possible loss of impregnant upon heating of iodized charcoals, completion of CH_3I trapping tests on charcoal samples from several sources, additional work on the determination of trace organic impurity levels at various ORNL reactor installations, as well as construction of experimental equipment for study of the efficiency of iodized charcoals for trapping elemental iodine vapor at high relative humidities.

Loss of Charcoal Impregnants Upon Heating

Observations made elsewhere showed that iodized charcoals may emit impregnant in the form of elemental iodine upon heating to temperatures well below their ignition temperature. This is possibly serious on two counts: (1) a possible decrease in effectiveness due to loss of impregnant, and (2) evolution of radioactivity if the charcoal had previously been exposed to radioactive methyl iodide or elemental iodine. Tests with iodized charcoal impregnated with lesser quantities of iodine (furnished by Barnebey-Cheney) are being conducted to determine their suitability for $\text{CH}_3^{131}\text{I}$ trapping and to determine their susceptibility to loss of impregnant upon heating. Test conditions were: air at 40 fpm, 70% relative humidity, and 1 mg CH_3I injected per g of charcoal. The results are contained in Table 1. According to these results, there is no significant difference in trapping capability of charcoals containing $\text{CH}_3^{131}\text{I}$ loading of 2, 3, or 5% of the particular impregnant that was employed. In addition we suspect that the charcoal containing 2% impregnant should be the less susceptible to loss of impregnant on heating since the initial impregnants are expected to be more tightly attached to the charcoal.

Table 1. Effect of Charcoal Impregnant Level
on Methyl Iodide Trapping Efficiency

Amount of Impregnant (wt. %)	CH_3 ^{131}I Removal Efficiency for a 2-inch Depth (%)
1	98.59
2	99.78
3	99.82
5 (present commercial product)	99.81

In a closely related study, iodized charcoals that had been utilized to exchange-trap CH_3 ^{131}I were heated at various temperatures, and the amounts of radioactivity evolved were determined. In the first experiment of this type, the behavior of two laboratory-prepared iodized charcoals was investigated. One was impregnated with 5.67% KI and the other with 5.00% (KI + I₂). The KI to I₂ molar ratio of the latter charcoal was 2 to 1. Thus, each of the charcoals was impregnated to the same concentration of exchangeable iodine. The following table shows the cumulative amounts of radioactivity evolved, relative to the original amounts, during heating periods of 4-hr duration at the indicated temperatures.

Table 2. Loss of Radioactivity Upon Heating of
Impregnated Charcoal for 4 hrs

Temperature, °C	Amount of Radioactivity Evolved (%)	
	KI-Impregnated	(KI + I ₂)-Impregnated
100	0.17	0.18
150	0.18	0.21
200	0.20	0.65
260 (1st)	20.6	35.4
260 (2nd)	37.8	46.3

As may be noted, only small amounts of radioactivity were evolved at 200°C and below, with the (KI + I₂)-impregnated material being the more susceptible to such loss. At 260°C, much larger amounts were evolved, with the (KI + I₂)-impregnated material again exhibiting a tendency to release its radioactivity more readily (as would be expected); however, the KI-impregnated charcoal also is fairly susceptible to such loss on repeated heating cycles at 260°C. Inspection of the distribution of the radioactivity collected downstream of the heated beds indicated that it was evolved mainly in the form of elemental iodine or, possibly, hydrogen iodide. Additional experimentation of this type is currently in progress, and further work on determining the optimum level of impregnation is planned.

Organic Impurities in Various Environments

As a part of the study of iodine behavior in containment atmospheres, an effort is being made to determine the levels of trace organic contaminants in various representative atmospheres. This information is being utilized in experimental studies of gas-phase reactions producing methyl iodide and also in the study of possible effects of accumulated organic material on the iodine and methyl iodide efficiencies of charcoal adsorbers operated continuously for extended periods of time. Samples were taken at the Bulk Shielding Reactor (BSR) at ORNL, the Health Physics Research Reactor (HPRR), located in an isolated, wooded area, and at the Carbide Recreation Park also in an isolated wooded area. Table 3 lists organic compounds identified and Table 4 lists the total hydrocarbons found and a breakdown into saturates and unsaturates. Generally, these samples as well as previous samples contained about the same organic compounds, but vary widely as to individual quantities and to total amounts.

¹R. E. Adams et al., ORNL Nuclear Research and Development Program Bi-monthly Report for November-December 1966, USAEC Report ORNL-TM-1742, January 13, 1967, pp. 16-18.

Table 3. Gas Chromatographic Determination
of Simple Paraffins and Olefins in
Air Samples (Parts per Billion)

Compound	Sample Location		
	Carbide Park	BSR	HPRR
Methane	36	56	45
Ethane	1	7	1
Ethylene	1	3	1
Propane	1	3	1
n-Butane	1	1	1
Propylene	-	1	-
Isopentane	1	1	-
n-Pentane	1	2	-
Pentene-1	4	-	-
n-Hexane	-	3	2
n-Heptane	-	8	-
	46	85	51

Table 4. Total Concentrations of Hydrocarbons Found
(Parts per Billion)

Sample	Total Unsaturated Hydrocarbons	Total Saturated Hydrocarbons	Total - All Hydrocarbons
Carbide Park	139	49	188
BSR	1185	62	1247
HPRR	146	67	213

1.5 PROPERTIES OF FISSION PRODUCT AEROSOLS PRODUCED BY OVERHEATED REACTOR FUELS

(AEC Activity 04 60 10 01 1)

G. W. Parker G. E. Creek W. J. Martin

Based on rate gas measurements^{1,2} almost exclusively, various estimates have been made of the extent of release from solid UO_2 of other volatile fission products expected from the various release processes which become controlling at increasing elevated temperatures. Since all of these are relatively slow, compared to the prompt recoil, allowance must be made for the steady-state radioactive half-life of each nuclide of importance. The five major processes for fission product migration to the void space are categorized as follows:

<u>Principal Release Process</u>	<u>Applicable Temperature Range</u>
I. Prompt Recoil	< 1000°C
II. Diffusion Through UO_2 Crystallite	1100 - 1650°
III. Equiaxed Grain Growth	1650 - 1900
IV. Columnar Grain Growth	1900 - 2800
V. Melting	> 2800°

In general, the release of a relatively long-lived volatile fission product is expected to be less than one percent by the recoil process, less than 20 percent by the diffusion process, less than 50 percent by equiaxed grain growth and above 50 percent by the columnar grain growth process.³

¹R. C. Nelson et al., Fission Gas Release from UO_2 Fuel Rods with Gross Central Melting I. Pellet Fuel, USAEC Report GEAP-4572, July 24, 1964.

²D. Davies, G. Long and W. P. Stanaway, The Emission of Volatile Fission Products from Uranium Dioxide, USAEC Report AERE-R-4342, June 1963.

³F. N. Nichols, Behavior of Gaseous Fission Products in Oxide Fuel Elements, USAEC Report WAPD-TM-570, Oct. 1966.

These estimates are based mainly on the so-called annealing type experiments such as reported at Second and Third Conferences on Nuclear Reactor Chemistry.^{4,5} However, since the nature of the original data was such that only the accumulated rare gas release was measured, it was necessary to re-examine the original data for the "steady-state" rate of release after the initial "burst". Typical steady-state release rate curves are reported by Booth and Rymer.⁶ In a later report⁷ further illustrations were present of the method of deriving steady-state, rare gas diffusion parameters for irradiated UO_2 .

By comparing the total release of rare gases with halogens observed in the annealing experiments⁵ it was concluded that the steady-state rate for iodine should be approximately twice that of the rare gases, although no quantitative measurements are reported for this observation because of the plate-out problem characteristic of iodine releases. However, a typical case in which the above data have been applied to a fuel operating temperature profile are given in Table 1. These estimates are assumed to represent that fraction of the ^{131}I iodine which would be available for release in the event of a cladding rupture accident under a condition which permitted complete volatilization of the steady-state void space accumulation. These values appear to be significantly higher than those currently expected according to recent reactor safety analyses.

⁴Oak Ridge National Laboratory, Nuclear Reactor Chemistry Second Conference, Gatlinburg, Tennessee, October 10-12, 1961, USAEC Report TID-7622, July 1962.

⁵Oak Ridge National Laboratory, The Third Conference on Nuclear Reactor Chemistry, Gatlinburg, Tennessee, October 9-11, 1962, USAEC Report TID-7641.

⁶A. H. Booth and G. T. Rymer, Determination of the Diffusion Constant of Fission Xenon in UO_2 Crystals and Sintered Compacts, Atomic Energy of Canada Ltd. Chalk River Project, Chalk River, Ont., August 1958.

⁷G. W. Parker, G. E. Creek and W. J. Martin, Fission Product Release from UO_2 by High Temperature Diffusion and Melting in Helium and Air, Oak Ridge National Laboratory (internal distribution only), USAEC Report CF-60-12-14, February 14, 1961.

Table 1. Method for Calculating Diffusion of Iodine-131 to Clad Gap Space

Zone		Rare Gas Released in 8 Days				
Boundary Temperature	Average Temperature	Volume Fraction	Percent Released		Total (percent x volume fraction)	
			1000 Mwd	4000 Mwd	1000 Mwd	4000 Mwd
Average Rod (BWR)						
1700	1675	0.04	4.8	16.1	0.19	0.64
1650	1575	0.12	2.4	11.8	0.29	1.42
1500	1410	0.20	1.0	6.6	0.20	1.32
1320	1210	0.28	0.3	2.5	0.84	0.7
1100	975	0.36	0.2	0.6	0.07	0.21
850						
Predicted percent of iodine in 8 days = rare gases x 2					1.59	4.29
					3.18	8.6
Leading Rod (BWR)						
2400	2326	0.04	36	72.1	1.4	2.88
2320	2215	0.12	17	58.1	2.0	6.97
2110	1925	0.20	8.2	31.2	1.6	6.24
1740	1486	0.28	1.4	8.3	0.4	2.32
1233	944	0.36	0.15	0.44	0.06	0.16
655						
Predicted percent of iodine in 8 days = rare gases x 2					5.46	18.57
					10.9	37.1

2.1 MODEL TESTS OF NATURAL TRANSPORT PHENOMENA

(AEC Activity 04 60 80 01 1)

L. F. Parsly

J. L. Wantland

J. K. Franzreb

The Nuclear Safety Pilot Plant is a facility for study of fission product behavior in a small (1350 ft³) containment vessel. Our objectives include proposing and testing mathematical models to account for fission product behavior in containment systems and study and evaluation of engineered safeguards. Our activities on the model test program have included the following:

1. Tests of the condensation model,
2. Small-scale flow studies of convective patterns in the MCV.

Condensation Model Tests

The results from run 18 were received and processed. The airborne iodine concentration vs. time data from May pack samples of the MCV atmosphere are presented in Fig. 2.1. The results have been normalized to make them easier to compare. The data indicate a fair agreement between runs 16 and 18 during the first 6 hours and considerable divergence after that. At 45 hours the normalized concentrations for the three runs are spread over a factor of 10. The data will be examined further in an effort to find the source of variation.

Run 19 was made on May 19, 1967. This was similar to runs 16-18 except that a fan was run in the MCV throughout the run. Our analytical laboratory has had an unusually heavy work load because of a scheduled shutdown of the Molten Salt Reactor Experiment and processing of the NSPP samples from run 19 is incomplete.

Small Scale Flow Studies of Convective Patterns

The 5°C Δt model apparatus has been completed and some shakedown operation conducted. This model vessel (12 in. OD x 18 in. high) is twice as large as the one previously used and is geometrically similar to it. Some changes have been made in the apparatus and instrumentation. The present model is completely submerged in the constant-temperature water

QRNL-DWG-67-7195

Fig. 2.1. Behavior of Airborne I_2 in the Model Containment Vessel NSPP.

bath and has a transparent lid. Previously the ink probes had been made from glass tubing drawn to a capillary and these probes were quite easily broken. An improved probe was made of stainless steel hypodermic tubing and difficulties are being encountered with this probe plugging. (The glass ones plugged too, but they seldom lasted long enough for this to be of concern.) Various methods have been used to control the flow of ink from the probe. Some techniques have been fairly successful but so far all have interfered with the movability of the probe. Work is continuing on this.

A hot plate for use in the MCV has been modified to be geometrically similar to the one in the model apparatus. It has a copper heating surface and two imbedded copper-constantan thermocouples. A temperature control device for this heat source is being adapted.

A computer program is being written to obtain numerical solutions to this type of flow problem. Difference forms of the equations of motion using the Boussinesg approximation are used in a modification of the Marker and Cell method devised by workers at the Los Alamos Scientific Laboratory.¹ The computational sections of this program have been completed and have been compiled. The major portions yet to be written are concerned with decisions on what output to be made and formats of output data.

¹J. E. Welch et al., THE MAC METHOD - A Computing Technique for Solving Viscous, Incompressible, Transient Fluid Flow Problems Involving Free Surfaces, USAEC Report LA-3425, Los Alamos Scientific Laboratory, March 21, 1966.

2.2 SPRAY AND POOL SUPPRESSION TECHNOLOGY PROGRAM

(AEC Activity 04 60 80 01 1)

T. H. Row, Coordinator

The scope of this program which had previously been directed to laboratory investigations of the mass transfer coefficient between gases (notably those containing CH_3I) and liquids (suitable for use as sprays), has been broadened to include scale model engineering tests of spray systems as well as the removal of fission product aerosols when contaminated gases are bubbled through suppression pools. We are also doing literature surveys and calculations to try to identify the conditions under which spray experiments should be run.

This program has been expanded to include the research necessary to give a complete evaluation of various solutions for use in removing fission products from containment systems. Included in the program will be a survey of the field of chemical additives to identify the most promising solutions for more extensive investigation. These solutions will be used as the basis for experiments in the single drop-wind tunnel which in turn will provide information for conducting experiments in larger facilities such as CRI, NSPP and CSE. A number of other facilities are also available for spray experiments. All solutions will be examined for thermal, chemical and radiation stability and corrosiveness.

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

A study of the transport of methyl iodide gas into a water drop suspended in a wind tunnel has been directed toward an examination of the effect of solution additives on this process. Experimentally, a concentration of $\text{CH}_3\text{I} \sim 10^{-5}$ millimoles per cc is present in the tunnel gases. The water drop under study remains suspended in the wind tunnel through the upward thrust of the flowing gas mixture.

It has been noted that penetration of CH_3I into the water drop as a function of time can be expressed by the following function:

$$\text{Log } F = C_1 + (\text{slope}) t, \quad (1)$$

where F is the fraction of the steady state concentration of iodide ion in the water phase at time t , and C_1 is an empirical constant with a value of approximately (~ -0.35 to -0.4). Equation (1) is an adequate empirical representation of the experimental data in the time interval above one second to the time of attainment of 90% saturation. It was observed that the slope of Eq. (1) could be made to yield a quantity with the dimensions of a mass transfer coefficient (velocity) if one assumed that the denominator of the slope involved the radius of the drop; i.e.,

$$\text{Slope} = \left(\frac{\text{velocity}}{r_{\text{drop}}} \right). \quad (2)$$

No claim is made that the slope is indeed dependent upon r to the first power, since division by r was introduced on the basis of dimensional considerations alone. However, we shall continue to use this assumption pending development of a better relationship.

In view of the general interest in $\text{Na}_2\text{S}_2\text{O}_3$ as an additive for reactor spray systems, we have examined (at a constant drop size of 0.2125 cm) the concentration dependence of the slope of Eq. (1) using Eq. (2) to calculate v_t for the $\text{Na}_2\text{S}_2\text{O}_3$ system (Fig. 1). The results were compared with those obtained using $(\text{NH}_4)_2\text{S}$.

The results show that v_t values with solution of $\text{Na}_2\text{S}_2\text{O}_3$ and $(\text{NH}_4)_2\text{S}$ (as well as one point for hydrazine) follow a common additive concentration dependency in the very dilute range with the v_t values rising from a limiting v_t value of 3.25×10^{-3} cm/sec for CH_3I in pure water. However, at additive concentrations of 0.3 wt % or larger, this is no longer true. The v_t values for $\text{Na}_2\text{S}_2\text{O}_3$ begin to gradually level off. At the higher concentrations the v_t values actually are depressed. The $(\text{NH}_4)_2\text{S}$, on the other hand, continues to rise sharply to a maximum with increasing additive concentration and falls off precipitously at higher concentrations. This behavior suggests that not only is $(\text{NH}_4)_2\text{S}$ comparatively more effective in facilitating mass transfer, but that a different trapping mechanism (mercaptan formation) may well be involved.

That two different mechanisms are involved is further suggested in the temperature dependence of the v_t values for $(\text{NH}_4)_2\text{S}$ as compared were those of $\text{Na}_2\text{S}_2\text{O}_3$. Preliminary unfinished measurements (not plotted) indicate that the v_t values of $(\text{NH}_4)_2\text{S}$ fall with increasing temperature whereas those of $\text{Na}_2\text{S}_2\text{O}_3$ increase with increasing temperature. The results at room temperature suggest that the initial rise in v_t of the additives studied point to a common effect on the liquid drop surface at low concentrations (less than 0.3 wt %). It may well be that a specific additive can be introduced whose properties are enhanced in the desired temperature range.

We shall now examine these conclusions in the context of the Griffith report¹ as it related to I_2 in water (Fig. 2.2). We begin by noting that the present v_t values for methyl iodide are lower than those of I_2 in water. The behavior is consistent with the fact that CH_3I deposits with greater difficulty than I_2 . As one begins to introduce $\text{Na}_2\text{S}_2\text{O}_3$ additive into water, the v_t values begin to rise (Fig. 2.2) until there occurs a leveling as the additive concentration reaches a value of 0.3 wt % (Fig. 2.1). Interestingly, the v_t value at which this takes place is practically identical to the mass transfer value of I_2 in water (Fig. 2.2). This suggests that a common chemical state (i.e., HI) is shared by both I_2 and CH_3I in H_2O .

Our suspicion that the HI species is shared by both CH_3I and I_2 is reinforced by an examination of the v_t values for $\text{Na}_2\text{S}_2\text{O}_3$ in the presence of 0.3% boric acid (Fig. 2.1) (the latter satisfying reactor safety requirements). In this case (Fig. 2.1), one notes that the addition of boric acid to $\text{Na}_2\text{S}_2\text{O}_3$ has reduced the mass transfer of methyl iodide to the limiting case of pure water alone. Conceivably, the formation of HI is an intermediate state for the removal of CH_3I from the drop surface to the interior. The addition of acid acts to repress the formation of this intermediate thereby leading to a reduction in the mass transfer coefficient of methyl iodide.

¹V. Griffiths, The Removal of Iodine from the Atmosphere by Sprays, UKAEA Report AHSB(S)R 45, 1963.

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Fig. 2.2. Comparison of Mass Transport Coefficients of ($I_2 + H_2O$) with ($CH_3 I + H_2O$).

This comparison of CH_3I behavior with that of I_2 , the latter described within the context of gas mass transfer considerations, serves to emphasize that the problem of understanding the role of additives on the mass transfer of gases into water drops is a tractable one. In addition it suggests that solution chemistry considerations affecting the removal of iodide into the interior of the water drop does play an important role in the overall mass transfer process.

This discussion has been essentially concerned with kinetic factors. The problem, however, of an effective spray additive is one that also is concerned with the affinity of the drop containing additive for the iodine species in the reactor containment vessel. This affinity is reflected in the liquid-gas distribution coefficients.

For the present, it is useful to note that the distribution coefficients of CH_3I in water containing $\text{Na}_2\text{S}_2\text{O}_3$ range from ~ 0.3 to 4.0 (depending upon concentration and temperature). On the other hand, those of $(\text{NH}_4)_2\text{S}$ range from 0.3 to 9.0 at 25°C . That there exist systems with affinities in water much higher (~ 130 as compared with 9) than those of $(\text{NH}_4)_2\text{S}$, but with less temperature sensitivity in v_t , is suggested in preliminary results obtained with 4 wt % tetraethylene pentamine. In addition, a favorable mass transfer coefficient of $\sim 4.3 \times 10^{-2}$ cm/sec at 26°C justifies a more detailed examination of amines, as well as other organic additives that have demonstrable affinity for the organic halides.

Spray Systems Analytical Model Studies (L. F. Parsly)

Analytical model studies were undertaken to obtain data for experiment planning purposes and constitute an extension of the analysis made by Griffiths¹ to the type of air-steam mixtures we anticipate would be produced by a loss-of-coolant accident in a water reactor system. We have written a program to calculate the properties and intermediate variables (viscosity, diffusivity, terminal velocity, etc.) needed and then to use these to calculate the gas-film deposition velocity. The calculations have been made for temperatures of 20 – 150°C and for drop sizes of 100 – 1600 μ . Some of the results obtained are presented in Tables 1 and 2.

Table 1 shows the calculated deposition velocities as a function of temperature and drop size. These results show that the deposition velocities decrease approximately a factor of 2 going from 30° to 150°C. This results principally from a decrease in diffusivity as the temperature increases. The diffusivity decreases because the model we assume takes a fixed volume of air at 30° and brings it to the final temperature and pressure by adding water vapor, so that the pressure increases rapidly with the temperature.

Table 2 shows the half-life for iodine removal by sprays assuming a constant liquid mass flow of $0.01 \text{ gcm}^{-2}\text{sec}^{-1}$ ($.147 \text{ gal ft}^{-2}\text{min}^{-1}$). (This is reasonable for the larger drop sizes but is probably unrealistically high for nozzles which would produce 100μ drops). The half-life is calculated by:

$$t_{1/2} = .693 / (v_g A)$$

where v_g is the deposition velocity in cm/sec and A is the interfacial contact area in cm^2/cm^3 . The contact area term depends on the liquid mass flow, the surface area per drop, and the terminal velocity. These factors interact in such a way that the contact area increases with temperature and the net effect is that the half-life is affected to a lesser extent than the deposition velocity.

The first spray run was made on June 20 and samples from it are now in process of analysis. Several modifications to the NSPP experimental facilities were necessary for the spray runs. These include addition of equipment for measuring gamma radiation in the liquid at the bottom of the containment vessel and for sampling this liquid remotely, addition of a Milton Ray pump to deliver spray at low flows, and, ultimately, replacement of the decontamination solution drain tank with a pressure vessel to permit operation at up to 50 psig.

Table 1. Deposition Velocity for Iodine to Sprays in Steam-Air as a Function of Temperature and Drop Size

Temperature °C	Drop Diameter, Microns					
	100	400	700	1000	1300	1600
	Deposition velocity cm/sec					
30	258	13.9	12.4	11.9	11.7	11.6
60	25.2	13.5	12.1	11.6	11.4	11.3
90	21.6	11.9	10.7	10.3	10.2	10.1
120	16.1	9.3	8.5	8.2	8.1	7.2
150	11.0	6.7	6.2	6.0	5.5	5.1

Table 2. Half-Life for Iodine Removal from Steam-Air by Sprays as a Function of Temperature and Drop Size

Temperature °C	Liquid Mass Flow = 0.01 gcm ⁻² sec ⁻¹					
	Drop Diameter, Microns					
	100	400	700	1000	1300	1600
Half life, sec						
30	0.13	4.7	17.6	39.6	70.4	111
60	0.13	4.6	17.2	38.3	68.5	108
90	0.14	4.9	17.9	40.1	70.9	112
120	0.16	5.5	19.9	44.0	78.3	108
150	0.19	6.3	22.4	49.0	77.5	113

2.3 HIGH EFFICIENCY AIR FILTER ENGINEERING MANUAL

(AEC Activity 04 60 80 01 1)

C. A. Burchsted

A. B. Fuller

The purpose of this project is to provide a guide for engineers and architects to the requirements for and design of air-filter systems which require the use of HEPA filters. Chapters 1 through 5 have been issued for review, and Chapter 6 will be issued shortly. Rewrite preparatory to final issue as an ORNL report will be started as soon as the remaining review comments have been received.

2.4 FILTRATION AND ADSORPTION TECHNOLOGY PROGRAM

(AEC Activity 04 60 80 01 1)

R. E. Adams J. S. Gill J. Truitt W. D. Yuille

This program is a part of the effort to advance the technology of fission product removal by filtration and adsorption techniques. The recirculating aerosol facility (RAF) is being utilized in a program of evaluation of the filtering efficiency of small samples of various commercial high-efficiency filter media toward the arc-generated aerosol consisting of oxides of uranium and stainless steel.

Previously it was reported that the presence of moisture seemed to play an important role in determining the efficiency of filter media and data at 33% relative humidity were listed.¹ Since that period a series of tests was initiated which, when completed, will produce information on filter efficiency at other humidities against the test aerosol after treatment under various conditions of concentration and aging. Summarizing the information available at this time one may say that there is no doubt that as the humidity of the system increases, the filtering efficiency decreases.

As part of our test series an effort was made to measure the reproducibility of the testing technique. A number of runs were carried out in which filter media from one source was used and aerosol concentration, age, humidity, flow velocity, and filter loading were all controlled as closely as possible. Random samples removed from sheets of the filter media proved to be rather uniform with a weight variation of $\pm 2\%$ for a given area. This series of tests, under $<15\%$ relative humidity, indicated an efficiency of 99.97% with a deviation of $\pm 0.013\%$, which is quite adequate for our purposes.

¹R. E. Adams et al., ORNL Nuclear Safety Research and Development Program Bimonthly Report for March-April 1967, p. 27, USAEC Report ORNL-TM-1864, Oak Ridge National Laboratory, May 10, 1967.

These observations seem to indicate that, under dry conditions, the filter efficiency for the oxide test aerosol is comparable to the efficiency for DOP aerosols; however, it should be noted that the measured efficiency for DOP aerosols is not suitable for predicting the filter efficiency to be expected for solid aerosols under highly humid conditions, such as might occur following a reactor accident.

A parallel study is being made to determine the effect of moisture on the behavior of the oxide aerosol and how this behavior, in turn, affects the efficiency of the filter media. Our findings indicate that the presence of moisture reduces the size of the aerosol agglomerates to perhaps an order of magnitude smaller than those produced under comparable conditions in a dry system. Further, the agglomerates seem to be more spherical in shape and there is little evidence of extended chains as noted under dry conditions.

Some effort has been made to study the effect of particle charge on the nature of agglomeration. The dry aerosol was passed over an alpha source (Polonium-210) in order to neutralize any charge on the aerosol particles by interaction with air ions produced by the alpha particles. This is a recognized technique in aerosol study. However, examination of these agglomerates revealed essentially no difference in the size and shape as compared to a similar aerosol agglomerate in the absence of alpha particles. Further investigation on the effect of particle charge and moisture on agglomeration is continuing.

2.5 SEPARATION OF NOBLE GASES FROM AIR BY PERMSELECTIVE MEMBRANE

(AEC Activity 04 60 80 01 1)

R. H. Rainey

Laboratory experiments are being made 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. ORNL values for the permeability of membrane supported on a Dacron backing were about half those for unsupported membranes previously measured by General Electric. However, the separation of krypton from oxygen (which largely determines the design of a processing cascade) was greater than previous studies had indicated. In an evaluation of the cost of a separations cascade, the lower permeability values resulted in an increase in membrane area required per stage, but the larger separation decreased the number of stages required.

Preliminary data from K-25 indicate that the permeability of the membrane to nitrogen is increased by about a factor of two by increasing the temperature of the gas from 25°C to 82°C. Studies of the effect of temperature on the krypton-oxygen separation factor are incomplete.

2.6 ANTISEISMIC DESIGN OF NUCLEAR FACILITIES

(AEC Activity 04 60 80 01 1)

O. H. Klepper R. N. Lyon

ORNL has been assigned the responsibility of 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 accomodate 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 accomodate more extensive ground displacement. An attempt will also be made to develop new conceptual designs within current or readily developed technology to accomodate ground displacement up to six feet, with simultaneous shaking and accelerations up to 1.0 g. As noted in the previous report, the firms of Burns and Roe, and Holmes and Narver had been selected and have initiated their studies. Additional study contracts are now in the final stages of negotiation with two other architecture-engineering firms.

The firm of Daniel, Mann, Johnson, and Mendenhall of Los Angeles, California will investigate floating concepts, and Kaiser Engineers will investigate the integration of a reactor plant into a single structural unit strong enough to allow vertical and horizontal displacement of the substrata.

R. N. Lyon attended the IAEA's Symposium on Aseismic Design of Nuclear Reactors held in Tokyo, Japan, June 12-17, where he and Dr. J. A. Lieberman, Assistant Director for Nuclear Safety of AEC Division of Reactor Development and Technology presented a paper about the earthquake engineering program of the AEC.

3.1 THE CONTAINMENT RESEARCH INSTALLATION

(AEC Activity 04 60 50 05 1)

G. W. Parker

W. J. Martin

N. R. Horton

The pre-operational shakedown tests involving release of radioiodine into various thermal and steam environments within the stainless steel liner of the CRI have been completed. CRI runs designated Nos. 103, 104, and 105 were conducted with 12 hr ^{130}I as the source activity to permit a more rapid experimental turn-around period. The runs were all completed successfully and no leakage from the tank was detected. A minor mechanical problem occurred in the failure of closure devices on the LOFT plate-out sampler. These particular ball sample units have been declared obsolete as continued mechanical problems seem to plague them. These three runs will be used to establish gross iodine behavior in CRI in preparation to continuing the CSE Fission Product Simulant Testing Program in this facility rather than in the smaller CMF. The result of the three runs show almost the expected behavior for I_2 and HI in air and steam. The half times for deposition were not characterized by a single rate but appeared to have a component of a quite short period (20 min) and an intermediate one of about 90 minutes and a very long one of some 30 to 40 hours. The measured methyl iodide production for each run gave values generally much less than 1%. An interesting anomaly occurred in run 104 in which after standing over night and perhaps reaching a dry surface condition a significant desorption from stainless steel was observed. The other runs did not show this different behavior. The data are compared in Table 1.

Table 1. Distribution of Iodine in CRI

	CRI-103	CRI-104	CRI-105
	Experimental Conditions		
Atmosphere	Air	Air-Steam	Air-Steam
Temperature at time zero	30°C	125.5°C	99°C
Pressure at time zero	28.8 psia	47.4 psia	51.5 psia
Aging time (hours)	20.4	20.9	21.7
Activity	^{130}I	^{130}I	^{130}HI
	Percent of Total Iodine		
Total in run	100	100	100
Plate-out on tank wall	77.7	84.1	86.8
Condensate	-	15.3	12.9
Activity from pressure release	8.5	.18	.068
Airborne activity after pressure release	13.4	.35	.06
Penetrating iodine	.06	.2	.02

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3.2 CHARACTERIZATION, CONTROL, AND SIMULATION OF FISSION PRODUCTS RELEASED UNDER LOFT CONDITIONS

(AEC Activity 04 60 50 05 1)

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

This program is concerned with the development and interpretation of the response of LOFT iodine-sampling devices under the environment expected in the LOFT meltdown experiment. 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 which 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.

Recently, a new approach to solution of this problem has been employed in which the difference in the rates of diffusion of the molecular iodine and particulate material forms the basis for separation. The sequence of components in the May pack was (1) a 2 in. long, 1.38 in. diameter, silver-plated cylindrical honeycomb with a 1/8-in. wide hexagonal channel, (2) three Gelman A high efficiency filters, (3) five ACG/B charcoal loaded filter papers, and (4) one 3/4-in. impregnated charcoal bed. Two tests with elemental iodine have been made; each test exposing duplicate May packs at 90% relative humidity and 90°C with a gas flow of 1 l/min. The distribution of the deposited iodine on the silvered honeycomb was determined by flattening it and cutting sections perpendicular to the direction of flow. Each of the sections was cut into three pieces to determine the radial distribution. In the first run 97.8% of the elemental iodine was trapped by the honeycomb. However, the radial distribution indicated that most of the iodine radioactivity was in a few channels about the center axis of the honeycomb. To equalize the gas flow in the honeycomb the opening in the inlet to the May pack was

increased from 1/4-in. to 1/2-in. and tapered out to the diameter of the honeycomb. The iodine distribution in the second test was found to be almost uniformly distributed radially, with 31% on the center third. As expected the elemental iodine retention was more complete (99.7%) than in the first test.

For successful application it must be demonstrated that the honeycomb cylinder does not trap a significant portion of the particles or particulate iodine. One test has been accomplished in an aerosol facility in which radioactive particulate aerosols were generated by arc-melting irradiated stainless steel tubes with a UO_2 insert. The aerosol was aged about 10 min and then passed through a honeycomb with a 15% relative humidity air flow of 1 l/min. Only 2.3% of the aerosol was deposited in the channels of the honeycomb. From the results of these tests a silver-plated honeycomb cylinder of the correct dimensions appears to be useful for analytical separation of elemental and particulate iodine. Further testing of this May pack sequence will be made under a wider range of conditions to establish its reliability.

The study of the reaction of elemental iodine with organic coatings of interest in the LOFT program was continued. The recommended Amercoat 66 system to be used on concrete surfaces consists of one coat 66 primer, two coats 66 surfacer and one coat 66 seal glass. Formation of organic iodides from this four-layer system has been reported previously. Examination of individual layers of the Amercoat system has been made by painting 1.2 liter stainless steel vessels and heating the sealed vessels at 100°C with and without the presence of elemental iodine. Gas samples were analyzed from the vessels with and without iodine by gas chromatography with dual detection (electron capture and flame ionization).

Amercoat 64 primer with iodine yielded ten major chromatographic peaks (electron capture detection), five of which were not found in the vessel without iodine. Four of the extra five peaks had the same retention time as methyl iodide, ethyl iodide, propyl iodide, and butyl iodide.

The vessels painted with two coats of Amercoat 66 surfacer and exposed to iodine yielded gas samples producing three extra chromatography peaks (by electron capture), one of which has the same retention

time as methyl iodide. The vessel with Amercoat 66 seal gloss single layer produced compounds which gave ten electron capture peaks, seven of which did not appear in gas samples from the vessel without iodine. Four of the extra peaks have the same retention time as methyl, ethyl, propyl and butyl iodides.

The five-coat Amercoat system commonly used for steel surfaces consists of: one coat Amercoat 64 primer, three coats Amercoat 66 epoxy and two coats Amercoat 66 seal gloss. This multilayer coating is being examined under the conditions described above and will be used for more detailed study of reaction variables such as iodine concentration and effect of moisture.

3.3 THE LOFT ASSISTANCE PARAMETRIC FISSION PRODUCT RELEASE PROGRAM (AEC Activity 04 60 50 05 1)

G. W. Parker

Two types of LOFT-Oriented Fission-Product Parametric Release Studies have continued according to the program proposed to simulate the core melting and redistribution model. The more basic study, the helium-blanketed induction heated tungsten crucible program has received an increase in interest because of the importance of the diffusion parameter to the calculation of plenum or clad gap activity. A recent report, WAPD-TM-570, "Behavior of Gaseous Fission Products in Oxide Fuel Elements", by F. A. Nichols (Oct. 1966) has given the first comprehensive theoretical treatment of this subject. In this report and in AERE-4342 by Davies, Long, and Stanaway mechanisms are proposed to account for the rate of emission of gaseous as well as other fission products and the correlation of fuel density, burnup, temperature spectrum, etc. The uncertainties associated with the calculation of plenum activity pointed out in the above documents has led us to conclude that each of these major parameters must be included in the planned study if a comprehensive model is to be developed. Accordingly, preparations for obtaining typical reactor grade fuel specimens covering the complete fuel density and burnup range (within the practical water reactor range) have been made. The following sources are under consideration:

<u>Reactor</u>	<u>Fuel Density (% theoretical)</u>
1. Dresden	93-94
2. CVTR	94-95
3. Saxton	95
4. SELNI	96

As soon as the basic studies are advanced far enough to indicate the relative significance of each parameter to the release model, it will then become somewhat simpler to choose only a few important specimens to test in the in-cell capsule-failure release apparatus which essentially assigns a transport value to the various regimes of fission product release.

4.1 IGNITION OF CHARCOAL ADSORBERS BY FISSION PRODUCT DECAY HEAT (AEC Activity 02 30 02 90)

C. E. Miller, Jr. R. P. Shields

The third in-pile experiment to measure the effect of fission products on the ignition temperature of charcoal adsorbers has been delayed to make a slight modification to include a second charcoal adsorber. The experimental unit will then have an adsorber in which heat will be applied to ignite the charcoal while fission gases and vapors are passing over it, as in the previous experiment and in addition a second adsorber with vacuum insulated walls to prevent loss of heat. The insulated adsorber will make it possible to measure directly the decay heat power from the fission products available in this experiment. Also, the temperature measurements throughout the bed and the radiochemical analysis following the experiment should give information regarding the fission product distribution throughout the depth of the charcoal bed.

The results of the decay heat power measurements and the fission product distribution in the charcoal adsorber will be used in a computer program to make transient heat calculations to determine the range of temperatures that fission product decay can produce in a commercial charcoal adsorber. To provide further input information to this program, equipment is being set up to make thermal conductivity measurements, on the various charcoals, in the temperature range ambient to 300°C.

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 purpose of this program is to examine in small-scale laboratory facilities the steam-carbon reaction concurrent with fission product release and transport that would occur due to accidental introductions of steam into the core regions of HTGR systems. Two high temperature furnaces with auxiliary equipment are being employed in these studies, one for the steam-graphite reaction, 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 at temperatures of 1000 to 1200°C using a steam-helium mixture having a partial pressure of steam of ~ 18 torr. The graphite container is 0.75-in. dia × 1-in. long and encloses a bonded bed of coated fuel particles 0.45-in. dia × 0.65-in. long. Reaction rates were determined at 1000, 1100, and 1200°C for one fuel segment which was oxidized to a final burnoff of 13 wt%. At this burnoff, oxidation appeared to have been largely confined to the graphite shell with little evidence of attack on the binder residue or filler in the bonded bed. The coatings on the fuel particles were unattacked. Solid cylinders of EGCR moderator graphite 0.75-in. dia × 1-in. long also were oxidized at comparable temperatures and partial pressure of steam. Reaction rates found for these cylinders agreed quite well with those measured for the fuel segments. Another fuel segment was oxidized at 1100°C for 25 hr at a partial pressure of steam of ~18 torr resulting in ~50 wt% burnoff. The graphite container was severely oxidized and the binder and filler completely oxidized. The coatings on the fuel particles showed only a limited attack with only a few failed coatings visible.

Hollow cylinders of EGCR moderator graphite 0.75-in. OD × 0.45-in. ID × 2-in. long were impregnated with barium nitrate solution containing

^{133}Ba . Following drying at 110°C , the cylinders were heat treated in dry flowing helium at 1200 and 1400°C . Counting data are incomplete but preliminary observations indicate that very little barium was transported by the dry helium at 1200°C . It appears, however, that a significant transport occurred at 1400°C .

Additional batches of pyrolytic-carbon coated fuel particles prepared at ORNL were oxidized at 1100 and 1200°C using helium-water vapor mixtures containing 1000 parts per million by volume of water vapor. As in the past, marked differences in reaction rates of various coatings with water vapor were observed but no satisfactory correlation of reactivity with physical properties of the coatings was possible.

5.2 HIGH-TEMPERATURE BEHAVIOR OF GAS-BORNE FISSION PRODUCTS

(AEC Activity 04 60 10 01 1)

A. P. Malinauskas

The primary objective of this research is to assess the effect of water vapor on the volatility and consequent migration of fission products released from HTGR reactor fuels. The initial phase of the work is concerned with an investigation of the compound TeO_2 .

Vapor pressure determinations are currently being performed with a transpiration apparatus, in which a carrier gas is saturated with the compound under study in one region of the apparatus (the reactor), and the compound transported in this manner to another region (the deposition zone) and deposited. The observed variation in the volatility of TeO_2 as a function of water vapor pressure is shown in Fig. 5.1. The data points represent measurements which were taken in the vicinity of 955°K and reduced to correspond to this temperature in an approximate manner; these data are therefore subject to additional correction, but the corrected values should not differ significantly from those presented.

A least squares analysis of the tentative data yields the following expression for the apparent vapor pressure $p_{\text{TeO}_2}^*$ of TeO_2 at 955°K .

$$p_{\text{TeO}_2}^* = (2.57 \times 10^{-4}) p_{\text{H}_2\text{O}} + 0.0262 ,$$

in which the pressures $p_{\text{TeO}_2}^*$ and $p_{\text{H}_2\text{O}}$ are given in torr. The corresponding equation which has been derived from the results of a previous worker¹ is given by

$$p_{\text{TeO}_2}^* = (2.98 \times 10^{-4}) p_{\text{H}_2\text{O}} + 0.0259 .$$

¹O. Glemser, R. V. Haeseler, and A. Muller, Z. Inorg. Allg. Chem., 359: 51 (1964).

Fig. 5.1. Apparent Vapor Pressure of TeO_2 at 955°K .

The two equations are in excellent agreement insofar as the vapor pressure of pure TeO_2 is concerned ($p_{\text{H}_2\text{O}} = 0$), but differ by about 15% on the magnitude of the effect of water vapor. This difference can be interpreted as a corresponding difference in the equilibrium constant for the reaction

$$K_p = \frac{p_{\text{TeO}(\text{OH})_2}}{p_{\text{H}_2\text{O}}} \quad .$$

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 experiment 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. Assembly of all components and check-out of the instrumentation and the control system was completed. We plan to start the operational checkout of the facility about July 5. This date is approximately 8 weeks behind the original schedule set forth for the project issued in February 1967. Many difficulties were encountered in the redesign of the facility for the high temperature operation, in the selection of materials of construction that would withstand the temperature (2800°F design criteria) in a 100% superheated steam atmosphere, and in the fabrication and assembly of the large number of precision and fragile parts. Several of the parts have required minor revision or redesign during assembly to facilitate proper fitting and to reduce stress concentration points throughout the temperature range of operation. During the initial assembly each part was carefully fitted to facilitate re-installation of the test section in a minimum down time after each test.

The first operational test will be an engineering test to determine the capabilities, limitations, and operational characteristics of the facility.

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 is designed to measure and characterize the release of non-gaseous 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, supplied by General Atomic is used for this and other nuclear safety studies to facilitate comparison of results.

The first poolside capsule, 06-11, was installed in the ORR May 3. The desired fuel graphite interface temperature of $1050^{\circ}C$ was achieved with a corresponding central fuel temperature of $1220^{\circ}C$. Operation was satisfactory until June 1 when a leak developed between the primary and secondary containment of the capsule. The leak was found to be sensitive to the temperature of the fission product sweep gas outlet line. It was necessary to turn off the trap and sweep gas line heaters, to retract the experiment to a position where the power is about 34% of normal and to schedule removal of the capsule from the reactor. Although the planned operation and test of this capsule could not be completed, approval of the local safety group was obtained for the conduct of a modified accident simulation test prior to capsule removal. Accordingly thirty-two hours before the reactor shutdown the capsule temperature was brought to "normal" conditions, $1050^{\circ}C$ in the graphite fuel interface, by changing the sweep gas to neon and advancing the capsule toward the reactor. Nine hours before the shutdown the capsule temperatures were raised to the planned accident conditions, $1450^{\circ}C$ in the graphite-fuel interface, ($1720^{\circ}C$ central fuel temperature) by advancing the capsule further toward the reactor. These conditions were held until the reactor shutdown. By this means it is hoped to salvage at least a portion of information from the experiment. The capsule was removed from the reactor June 16. Post-irradiation examination and analysis of non-gaseous fission product release is in progress. Table 1 summarizes the fission gas release data taken during the irradiation. The release rate during the nine hour simulated accident had not reached equilibrium.

Table 1. Fission Gas Release From HTGR Nuclear Safety Test -
ORR Poolside Capsule

Date of Sample	Temperature, °C		R/B Ratio			
	Fuel	Graphite ^a	Kr ^{85m}	Kr ⁸⁷	Kr ⁸⁸	Xe ¹³⁵
5-17-67 ^b	1230	1050	9.13×10^{-5}	1.05×10^{-4}	1.51×10^{-4}	2.66×10^{-5}
6-15-67 ^c	568	460	3.57×10^{-5}	3.58×10^{-5}	1.08×10^{-4}	1.81×10^{-5}
6-15-67 ^d	1720	1450	9.65×10^{-5}	-	6.39×10^{-4}	2.37×10^{-5}
6-15-67 ^e	1720	1450	8.81×10^{-4}	4.64×10^{-3}	1.34×10^{-3}	3.77×10^{-3}

^aEstimated fuel-graphite interface temperature.

^bFission gas release with experiment at "normal" operating condition.

^cGas sample taken immediately before simulated accident.

^dGas sample taken after 4.5 hours operation at simulated accident conditions.

^eGas sample taken after 6.5 hours operation at simulated accident condition.

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

(AEC Activity 04 60 10 01 1)

C. E. Miller, Jr. S. H. Freid B. F. Roberts

The first in-pile experiment with a HTGR type fuel under accident conditions has been performed. The fuel, provided by General Atomic, was pyrolytic graphite coated (U, Th)C₂ and ThC₂ spheres imbedded in a resin binder and contained in a graphite can. Moist helium was passed over the fuel specimen, which was maintained at 1050°C, for a total of 52.5 hours after which the experiment was disassembled in the hot cells. Throughout the experiment the CO concentration was continuously monitored using an infrared analyzer.

Several effects were apparent: (1) The steam graphite reaction at this temperature is very fast as indicated by a very sharp rise in the CO concentration when the helium stream was moistened. (2) At a normal flow rate of 1400 cc/min the CO concentration initially rises to approximately 1000 ppm and increases steadily throughout the experiment. (3) The post-irradiation examination revealed that most of the graphite can and all the resin binder had reacted to form carbon oxides leaving the fuel spheres loose in the furnace. (4) The fuel spheres were found to have a significant number of unclad and partially clad fuel cores. (5) In addition, preliminary activity surveys indicated that at the temperature of this experiment a limited amount of fission product release other than Xe and Kr occurred.

A portion of the fuel is now being examined metallographically and the experiment is also undergoing radiochemical analysis.

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

G. D. Whitman

A preliminary edition of the report, "Technology of Steel Pressure Vessels for Water-Cooled Nuclear Reactors," ORNL-NSIC-21, was issued on May 19, 1967, to selected reviewers for final comment prior to publication. An index is being prepared for inclusion in the final version and additional comments are to be considered for inclusion in the report if received by July 7, 1967.

6.2 EXPERIMENTAL AND ANALYTICAL INVESTIGATIONS OF NOZZLES

(AEC Activity 04 60 70 01 1)

B. L. Greenstreet R. C. Gwaltney

The experimental investigation of nozzles radially and nonradially attached to spherical shells and flat plates is 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 radially attached 7 7/8 in. OD nozzle on the first model has been bored out in preparation for the next series of tests with the thinner wall. The second model with a nozzle attached at an angle of 22 1/2 degrees is ready for testing.

The models for the flat plate studies at the University of Tennessee are presently being machined. The standard plate has been strain gaged and tested. Both uniaxial and biaxial tension tests were performed. The outer gages were 7 in. from the center of the plate. All gage readings agreed within 5%. It has become evident that a separate loading frame will be necessary to hold the plates when external loads are applied to the nozzles. The present biaxial tension frame will not support the edge of the plates for out-of-plane loads. The two loading frames will also allow two tests to be done simultaneously.

Fabrication of the second spherical model containing a series of cluster holes is continuing at Auburn University. The loading frame and pressurizing system have been fabricated. Preliminary tests on the first model containing the large radial nozzles have begun. A computer program for reducing the data has been written.

The analytical studies being done by Applied Technology Associates on nozzles attached in clusters have led to a ligament method of analysis

for a spherical shell perforated by many radial holes. A report has been written.¹

¹Robert Rung, V. L. Salerno, and J. B. Mahoney, Effective Elastic Coefficients for a Spherical Shell Perforated by Many Radial Holes, Applied Technology Report No. 112-A, May 1967.

6.3 VARIABILITY OF IMPACT TEST DATA FOR LOW ALLOY STEELS

(AEC Activity 04 60 70 01 1)

R. G. Berggren

J. T. Stanley

The purpose of this program is to determine the extent and cause of scatter in Charpy V-notch impact data and the relationship between Charpy V-notch, Drop Weight NDT, and yield strength for several pressure vessel steels. On July 1, 1967, this program will become an integral part of the Heavy Section Steel Technology (HSST) Program (see Section 6.4 of this report).

In preparation for the HSST program as well as for this program two 12-inch thick plates of ASTM A-533 Grade B, Class I steel were purchased. These plates each weigh more than 50 tons and are ten-feet wide by over 20-feet long. (Two additional plates were purchased under the HSST program.) The first 12-inch thick plate has now been quenched and tempered, final nondestructive inspection and flame cutting has been completed. Stress relieving will be completed about June 30. A two-feet by plate width section from the midlength of the plate will be used for the initial "variability" studies on this plate. Tensile tests, Charpy V-notch impact tests, Drop Weight tests, hardness traverses, and metallography will be conducted on material remote from the plate edge and from surface, 1/4 thickness, and center of the plate. Tensile tests will be conducted at several temperatures from room temperature to 700°F. Drop Weight tests will be conducted using various specimens (e.g. P-1, P-2 and P-3 as defined by ASTM-E208). Type P-1 Drop Weight specimens will have the long dimension perpendicular to the principal rolling direction. All other tests, including Charpy V-notch tests, will be conducted on specimens oriented both with and transverse to the principal rolling direction. Additional tensile and Charpy V-notch tests will be conducted on material from several other depths in the plate as well as for varied distances from the quenched edge of the plate. Fourteen inches was cut from the remaining quenched edges of the plate both to eliminate edge effects in plate specimens for the HSST program and to provide material for weldments in the HSST program. Six specimens, 15 inches by 90 inches by plate thickness

(12 inches), for Drop Weight Tear Tests to be conducted by the U.S. Naval Research Laboratory were cut from one of the plate sections. Subsequent to the Drop Weight Tear Tests, portions of these specimens will be utilized in the variability study.

The second 12-inch plate has been ultrasonically inspected prior to heat treatment and heat treatment for hot forming has been simulated.

A six-inch thick plate of Armco SSS-100 steel is now on hand. This plate is a quenched and tempered alloy steel covered by ASTM Standard A-517, Grade E. The Armco Steel Corporation is also preparing specimen weldments for this program. The test program on this steel should be finalized shortly. Test results to be obtained by the Armco Steel Corporation on one portion of this plate will be available around the end of July.

Installation and testing of the Drop Weight Test facility has been completed. Normalization procedures for the ASTM A-533 Grade B, Class I steel will, of necessity, be conducted after receipt of the 12-inch plate sections. Parts for the specimen conditioner for the Charpy Impact Test machine have been completed, and installation should be completed early in July.

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

F. J. Witt

The Heavy Section Steel Technology Program is aimed at investigating the effects of flaws, variation of properties, stress raisers, and residual stresses on the strength and structural reliability of present and contemplated water-cooled reactor pressure vessels. The program is divided into eleven tasks only seven of which are currently being pursued. The other four tasks are in various stages of planning. Work on three of the four tasks will begin during the first quarter of FY 1968.

Four 10 feet by 20 feet by 12 inch plates have been purchased from Lukens Steel Company. A picture of one ingot being placed on the rolling mill in preparation for rolling one of the plates is shown in Fig. 1. The ingot is around 2100°F and weighs in the order of 110,000 pounds. One of the plates has been heat treated to ASTM A 533 Grade B, Class 1 steel by Combustion Engineering, Inc. Contracts have been let for similar treatment of two other plates. The fourth plate will be used in the heat treatment and rolling reduction studies. A contract is being negotiated for this work.

The plate which has been heat treated by Combustion Engineering was instrumented with thermocouples as shown in Figure 2. The thermocouple leads are encased in stainless steel tubing as seen in the figure so as to assure that the leads are not shorted out during the quenching procedure. The plate being removed from the trolley immediately prior to quenching is shown in Fig. 3. The plate is at 1600°F. Ultrasonic examinations were made prior to and following the heat treatment. These results are currently being tabulated and compared. The various specimens to be taken from this plate and the tests to be run on each were described in the preceding sections (6.3 Variability of Impact Test Data for Low Alloy Steels) which as of July 1, 1967 will become an integral part of the HSST program.

The simulated service test includes the testing of a 10 foot diameter, 12 inch thick pressure vessel with a cylindrical test section of thinner

Photo 88008

Fig. 1. A 55 Ton Ingot Being Placed on Rolling Mill in Preparation for Rolling.

Fig. 2. A Plate Instrumented with Thermocouples Prior to Heat Treatment.

Fig. 3. A Plate at 1600°F Being Moved to the Quenching Tank.

material. Tests will be performed over a range of temperature from the nil-ductility-temperature (NDT) of plate material to the operating temperature of light water reactors (around 550°F). A conceptual drawing of the vessel is shown in Fig. 4. The vessel is some 40 feet long and weighs around 600,000 pounds. The heat transfer and pressurizing systems are currently in the conceptual design stage. Detailed analyses are also under way in several areas.

A semi-annual information meeting was held at Oak Ridge National Laboratory on June 6, 1967. Some fifty-five visitors attended the meeting which included a summary of related industrial and AEC programs. A progress report is scheduled to be written by August 15 which will summarize in detail the activities of the HSST program.

SIMULATED SERVICE TEST VESSEL

Photo 67-711

UNION
CARBIDE

Fig. 4. A Conceptual Drawing of the Simulated Service Test Vessel.

6.5 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 has defined a research program for the development of stress indices for piping, pump, and valve components at the request of the various code writing bodies. The PVRC program consists of 12 individual task items outlining analytical and experimental work necessary to define stress indices for the various components covered by the respective codes. In anticipation of the PVRC program becoming a jointly sponsored AEC-Industry effort, work has begun either at ORNL or under subcontract to ORNL on several of the tasks. These are PVRC Task No. 2, "The Evaluation of Existing Data and the Outline of Additional Data Requirements for Elbows, Curved Pipe, and Miter Bends"; PVRC Task No. 3, "The Evaluation of Existing Data on Elastic Stresses and Flexibility of Specified Piping Components"; and PVRC Task No. 6, "The Development of Stress Indices for Specified Axisymmetric Piping Components". A parameter study is underway to define the stress indices for welding elbows and curved pipes as a function of dimensions, applied moments, and internal pressure, based on theoretical studies in the literature. These indices will be included in the ASA B31.7 piping code and in the AEC-RDT standards currently being written.

We have also begun accumulating, abstracting, and cataloguing available literature which is pertinent to the general problem of stress indices for piping components, valves, and pumps. The literature survey has revealed that industry-wide dimensional standards for commercial pipe fittings such as tees, reducers, and threaded and socket welded pipe fittings do not exist. In order to properly relate the results from theoretical stress analyses to such standard components and to develop adequate stress index values, dimensional information is needed. We therefore purchased a number of carbon steel and stainless steel fittings from several manufacturers for detailed dimensional inspection and comparison. Some of the components which have been purchased will be available for strain gaging and testing. We are also attempting to obtain dimensional information from the manufacturers.

7.1 HTGR SAFETY PROGRAM OFFICE

(AEC Activity 04 60 70 01 1)

S. I. Kaplan E. A. Nephew
A. E. Goldman M. D. Silverman
 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 effects on HTGR safety was begun.

High-Temperature Physics

A subcontract with Pacific Northwest Laboratory (PNL) involving the use of the High Temperature Lattice Test Reactor (HTLTR) as a transient test facility for HTGR fuel elements was prepared and sent to AEC for approval. The work to be performed under this contract will be to determine the feasibility of using the HTLTR as a transient facility, to provide a preliminary safeguards analysis, and to prepare preliminary designs for any reactor modifications, including a cost estimate of these modifications.

We are also, independently, conducting exploratory calculations to assess the feasibility of improving the pulse characteristics of transient in UHTREX by varying the uranium-to-thorium loading. We have calculated the neutron generation time as a function of uranium loading and are developing a capability for performing point kinetics calculations.

Coolant Chemistry

A member of the staff attended the first information meeting of the Graphite Coordination Working Group at Hanford, Washington. Research and development people from industry and the national laboratories discussed in depth studies in the field of steam-graphite reactions and related subjects such as carbon deposition, radiolytic effects and radiation-induced reactions, boronated graphite, and the use of electron microscopy as an analytical device for studying these reactions. A summary of research requirements for HTGRs was also a topic of considerable discussion.

Systems and Components

An analysis of HTGR systems and components, directed toward considering whether the malfunction or maloperation of various inter-related reactor systems and components can directly or indirectly cause or seriously intensify a reactor accident, was started.

Supplemental Activities

The Program Office participated in a briefing session on HTGR Safety Research for the Safety Research Subcommittee of the ACRS, held in Washington on May 16. Material presented by the Program Office Director included a summation of HTGR safety-related research currently in progress, and a description of areas where it is felt that information is lacking.

The Program Office Director also attended the ACRS Subcommittee preliminary hearing on the Fort St. Vrain Nuclear Generating Station, on June 12 in LaJolla, California.

7.2 NUCLEAR SAFETY INFORMATION CENTER

(AEC Activity 04 60 70 01 1)

J. R. Buchanan Wm. B. Cottrell

A comprehensive indexed bibliography on the transportation and handling of radioactive materials was issued in June as ORNL-NSIC-33.¹ The bibliography of about 700 references was prepared in conjunction with an ORNL group that has been asked by the USAEC to develop a criteria against which newly designed shipping containers may be judged. The primary basis of the cask criteria are the AEC regulations governing the transportation of radioactive materials, data developed from experimental programs, defensible theoretical treatments, and good engineering judgment. One of the first tasks was to carry out this literature survey. The resulting information was judged to be of such interest to all persons concerned with shipping and storage of radioactive materials as to warrant publication.

The Center's ninth quarterly bibliography of accessions was distributed in May.² It contained 951 references sorted into NSIC's 19 subject categories.

NSIC includes in its coverage available reports from outside the USA and continues efforts to see that this coverage is broadened. During the report period, two staff members conferred with officials of several international technical information handling organizations. Celia Murphy, NSIC Chief Information Specialist visited the Centre d' Etudes Nucleaire, Saclay, France and Gmelin Institute, Frankfurt, Germany; and E. N. Cramer, NSIC staff specialist on reactor operations stopped at the international Atomic Energy Agency, Vienna, Austria; EURATOM, Brussels, Belgium and the Committee on Reactor Safety Technology, Paris, France. Mr. Cramer was

¹L. B. Shappert and R. S. Burns, Indexed Bibliography on Transportation and Handling of Radioactive Materials, USAEC Report ORNL-NSIC-33, June 1967.

²NSIC Staff, Indexed Bibliography of Current Nuclear Safety Literature-9, USAEC Report ORNL-NSIC-34, May 1967.

returning from an assignment to the Thailand Office of Atomic Energy for Peace for 1 month to assist in training reactor supervisors for their 1Mw research reactor.

During May and June, NSIC received and answered 97 information requests. There were 16 visits to the Center for staff consultation.

7.3 DISCUSSION PAPERS ON VARIOUS ASPECTS OF
WATER COOLED REACTOR SAFETY
(AEC Activity 04 60 80 03 1)

J. W. Michel

A draft of all discussion papers incorporating internal ORNL review only has been completed and sent out for detailed review by the Commission and the nuclear community. The draft copies were sent to the AEC, the four nuclear suppliers (G. E., Westinghouse, Combustion Engineering and Babcock and Wilcox), the major architect-engineers, some of the reactor operators and selected individuals at the National Laboratories and Universities. External review committees have met and reviewed two of the papers, and review of the others are scheduled in July and August. The final report of all studies will be issued as expeditiously as possible following the meeting of the external review committees. It is expected that two to three months will be required after the external review meeting for revision, editing and printing. The actual or scheduled date of the review meeting for each of the papers is listed below.

Table 1. Review Date for the Nuclear Safety Discussion Papers

Paper	External Review Meeting
Missile Generation and Protection in Light-Water Reactors	August 67 (tentative)
Potential Metal-Water Reactions in Light-Water Power Reactors	June 22, 1967
Emergency Core Cooling Systems for Light-Water Power Reactors	July 11, 1967
Air Cleaning as an Engineered Safeguard in Light-Water Power Reactors	August 67 (tentative)
Containment System Testing	August 67 (tentative)
Earthquakes and Power Reactors	July 5-14, 1967
Reactor Protection Instrument Systems	August 1, 1967

7.4 FUEL TRANSPORT SAFETY STUDIES

(AEC Activity 04 60 80 03 1)

L. B. Shappert

The purpose of this program is to develop a code which sets 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 of casks. In addition to the code, a backup document is being prepared which 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. More recently members of the ORNL program have participated in AEC review of current shipping cask designs developed by various manufacturers.

Review of Current Designs

The AEC established a special task group including ORNL, to undertake a safety analysis of the Elk River spent fuel shipping cask. The task group's report was submitted to Chicago Operations Office on June 30, 1967. This analysis considers not only the AEC and ICC requirements, but also the IAEA regulations.

The ORNL project was also asked by COO to review the technical support document for the Atomics-International-designed cask for use in the decommissioning of Hallam reactor and to make recommendations regarding fabrication problems which have arisen. Review of the document has been submitted to the AEC and we expect to have recommendations regarding the cask fabrication shortly. The problem concerns the bonding between the outer steel shell and the lead shielding which was presumed in the cask safety analysis but is not guaranteed by the fabricator. If the cask cannot be shown to be adequate without bonding, some other method (such as a model test) will have to be used to indicate that the unbonded cask complies with the regulations.

Structural Integrity

In recent weeks there has been considerable interest in concrete-shielded casks for the shipment of neutron sources. Since no adequate analytical means exist for determining the damage resulting from a 40-in. drop onto a piston, an experimental program has been initiated using the model drop frame located in Building 1000. Tests will be conducted using $2 \times 4 \times 8$ in. cast limonite concrete bricks to investigate this damage phenomena.

A second test program to support the effort of the Paducah depleted uranium shielded shipping cask program will also utilize this drop frame. Four bricks of depleted uranium were received from Paducah and will be tested to determine the damage resulting from piston impact.

Criticality

A draft of the criticality chapter of the Criteria has been completed and is presently being edited prior to final inclusion of Vol. 1.

Uranium Cask Work

A program to design, build, and test a demonstration fuel element shipping cask made from laminated uranium plate is being conducted by Union Carbide at Paducah. The cask design is complete and shop fabrication prints have been made. A program of tests and demonstrations has been formulated and detailed plans for the individual tests are being developed. The prints and overall testing program were reviewed and approved by AEC personnel at a meeting at ORNL on May 4, 1967. All uranium and other materials required to build the cask have been received and fabrication was started during the latter part of the month. A program to collect detailed fabrication costs was established so that such cost engineering factors as the cost per foot of weld, the cost to form plate, etc., can be determined for future extrapolation to other cask designs.

Several of the fabrication procedures planned for this cask were used in the building of a large radioisotope shipping cask for ORNL and no difficulties were encountered on that job. These procedures included: heat

treating the final assembly, rather than the individual plates, to randomize grain and prevent thermal cycling growth; the use of a plasma sprayed tungsten diffusion barrier to prevent low melting alloy formation between uranium and stainless steel cladding during a fire; and the use of a silicone coating to permit heating of uranium plate in an open furnace prior to its being sheared or formed into cylindrical laminations.

Heat Transfer

The heat transfer portions of Volume 1 and 2 including Decay Heat Load, Solar Heat Load, as well as the Fuel Magazine Design Section, have been completed. Programing and debugging of equations for internal heat transfer and fire analysis are progressing.

Fire Analysis. The fire analysis program is specifically for cylindrical casks which are either unfinned or finned externally. The program is completely general in that it requires only minimal input data such as thermophysical properties (of the materials of construction of the cask) and the cask geometry. A nodal heat transfer analysis of the example cask from Vol. 1 of the Acceptance Criteria has been carried out using an existing computer code. The analysis predicts that melting of the lead in the example cask sides will begin at 0.4 to 0.445 hr* after commencement of a 1475°F fire as compared to 0.504 hrs as predicted from a hand calculation. This is a close check when one considers the degree of accuracy which is realistically possible with hand calculations. We plan to use the aforementioned independent solutions for the example cask of Vol. 1 (i.e., the hand calculation and the nodal analysis) to evaluate and gain confidence in the results of the fire analysis computer program being presently developed. A final report on the computer program development along with data concerning the comparisons which are made will be submitted next month.

*A time range is given because the nodal analysis code does not account for phase changes and thus makes the exact time of melting quite difficult to determine.

Internal Heat Transfer. First results of the model simplification for internal heat transfer have been obtained. The method of approach upon which the results are based was to analyze an assumed, simplified model and compare the results with data obtained earlier, more rigorous analyses using the THT computer code (i.e., the six detailed temperature mappings shown in Vol. 1 of the Cask Acceptance Criteria Manual).

The maximum "error" involved for any of the three heat flux distribution conditions occurs for the uniform one and is about 100°F too high at 1300°F. Closer agreement may be achieved by further revision of the simplified model. One such revision would be to include the wrapper nodes and use heat-generating lumped fuel strips within the wrapper to replace the fuel pins. This model would require knowledge of an effective thermal conductivity of the lumped fuel strips.

Further effort on the simplified model analysis has been deferred until next fiscal year at which time it is hoped that a simplified model may evolve which does not require use of the THT code. All effort is now being concentrated on obtaining additional thermal profile maps of internal cask structures. A more detailed description of the simplified thermal model along with the additional temperature maps will be written up by July 1.

7.5 COMPUTER HANDLING OF REACTOR DATA FOR SAFETY (CHORDS)

(AEC Activity 04 60 70 01 1)

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

Preparation neared completion for initial computer retrieval of safety-relevant information about the design of nuclear power plants to assist the AEC Division of Reactor Licensing in their evaluations. Work is over 95% complete on the identification and listing for computer input of design characteristics of water-cooled power reactors in the areas of containment, safety controls, design basis accident, and containment cooling system. This material comprises 8000 items on 430 pages. Data for various parts of the first seven reactors is 15 to 100% complete. Computer programs for input and initial output are being debugged. Initial operation is planned for early August 1967. Work is underway to provide for computer storage of operating experience data received by the AEC Divisions of Compliance and of Operational Safety. This data file will be cross referenced to the file of design data to avoid unnecessary duplication.

Future work is planned to extend coverage to design data for four additional systems, and three additional reactors. Initial emphasis is being placed on the identification of the kinds of operating data required for reliability analysis, and on the definition of systems and classes of components to be given priority.

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

R. A. Schmidt

The RDT standards program was instituted to provide a foundation on which to base the design, procurement, fabrication, erection, and operation of government-owned or -sponsored reactors with the assurance that a safe, reliable, high-quality installation will be achieved. The standards will call on numerous sources of proven technology and current experience to provide a means to attain successful reactor installations.

A program proposal was submitted to AEC which presented a breakdown of standards, manpower requirements, schedules, and engineering sub-contracts. The scope, staffing, and scheduling of the program will be more clearly established after review of the proposal by the Division of Reactor Development and Technology.

Approximately 50 of the 75 standards provided by DRDT have been updated and submitted for comment. Seven of these have been approved for distribution to AEC offices and contractors as tentative standards.

Working relationships have been established with several installations to expedite the interchange of information necessary to meet current needs for RDT programs. Groundwork has been laid to incorporate standards from sources outside ORNL into the RDT standards program, and to consolidate similar standards from several sources into a single document where possible. Efforts are being made to establish a format or formats and a single, comprehensive numbering system for use by all participants in this program.

Consideration is being given to the application of computer storage and retrieval techniques for pertinent information in each standard. Keying the standards to computer programming will provide methods for exchange of information among other activities, such as CHORDS, NSIC, LMIC, and updating of national codes and standards.

A summary of the Nuclear Instrument Module (NIM) standard survey was prepared and submitted to DRDT. It was the consensus of most users and manufacturers of reactor instrumentation that although standardization

has many advantages, the NIM standard, which is successfully applied in nuclear research instrumentation, does not satisfy the diverse needs of reactor instrumentation.

ORNL was requested to investigate activities of IEEE and other national and international standards groups as sources or bases for reactor protection systems standards. It is expected that a series of reports will be prepared on this subject as the respective task forces develop information, codes, or standards in this field.

United Nuclear Corporation, under a subcontract to ORNL, is conducting a survey and preparing a classification for components of BWR and PWR primary systems. The survey will include data and information from several current generation reactors which are now under design or construction. A second task undertaken by United Nuclear Corporation is the review, evaluation, and recommendation for updating and expanding a vessel design manual.

8.1 Technical Progress Review Nuclear Safety
(AEC Activity 07 13 02 02)

J. P. Blakely Wm. B. Cottrell

Nuclear Safety is a Technical Progress Review prepared and edited at ORNL under contract to the USAEC. On June 23 the Bureau of the Budget approved the Nuclear Safety-DTI request for bi-monthly publication of the journal. Copy flow preparation for the Summer 1967 issue, Nuclear Safety 8(4), was completed at DTIE and sent to Germantown on the same date. This issue will then be transmitted immediately to the Government Printing Office and will be included in the count for FY 67. It contains 136 pages, and brings the fiscal year total to 584 in five issues.

Comments are presently being received from AEC and others on Nuclear Safety 8(5), which has been under review in draft as ORNL-TM-1847 since May 19. This issue presently contains articles by ten authors from outside ORNL and by five authors from within the Laboratory. This will be the first issue with a bi-monthly designation, and will be dated Sept.-Oct. 1967; publication date is anticipated around September 1.

The Nov.-Dec. 1967 issue, Nuclear Safety 8(6), was distributed for review and comment on July 3 as ORNL-TM-1873. It contains a lead article by F. R. Farmer (UKAEA) concerning his interesting reactor siting criteria, and other articles representing the work of eight authors from outside ORNL and seven from within. Review of the draft is scheduled for completion in early August, with publication around November 1.

Material is presently being received for Nuclear Safety 9(1), Jan.-Feb. 1968. A Feature Article by Edward Teller on Urban Reactor Siting has been received for this issue. The draft is scheduled for review beginning in late August, followed by publication around Jan. 1, 1968.

A limited-contact (AEC personnel and contractors) survey was made in March by NSIC regarding Nuclear Safety. In the 60-odd responses the following comments were made:

Degree of Usefulness

High - 23 Fair - 35 Low - 1

Depth of Coverage

Too General - 0 Too Specific - 1 About Right - 56

Quality of Technical Content

Excellent - 10 Above Average - 33 Average - 15 Poor - 0

Utilization

Read most of the articles - 9

Read a few articles after checking abstracts - 49

Put on Reference Shelf - 18

The several volunteered suggestions for improved timeliness and accelerated publications will be responded to in the upcoming bi-monthly schedule.

INTERNAL DISTRIBUTION

1. R. D. Ackley	53. W. R. Grimes
2. R. E. Adams	54. R. C. Gwaltney
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