

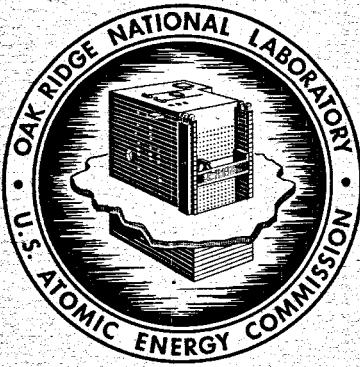
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ORNL-4396

UC-80 – Reactor Technology

MOLTEN-SALT REACTOR PROGRAM
SEMIANNUAL PROGRESS REPORT
FOR PERIOD ENDING FEBRUARY 28, 1969



OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

Outside the more active core region there is an undermoderated region containing 37% by volume of salt and designated as zone II. This region reduces the fissioning, and therefore the neutron leakage from the core, by increasing the fraction of absorptions in the thorium. We reduced the thickness of the undermoderated portion to 15 in., and rather than using graphite spheres, we now provide 2-in. by 10-in. by 14-ft-long graphite slabs with flow passages between them to provide the specified salt volume and for cooling.

Between the undermoderated region and the reflector graphite there is a 2-in.-wide annular space filled entirely with fuel salt which accommodates relative

dimensional changes, provides clearance for removing and replacing the reactor core assembly, and further reduces the neutron leakage. The width of this space is a compromise with the fuel-salt inventory.

The reflector region consists of 12- by 30- by 48-in.-high graphite blocks stacked around the removable core assembly. As with the core elements, the buoyant force of this graphite in the fuel salt is normally restrained by the top head structure. When the reactor is empty of salt, the weight is carried by the lower head. There is a $\frac{1}{4}$ -in. annular clearance between the reflector graphite and the vessel wall to permit an upward flow of fuel salt to cool the wall and maintain it below the design temperature of 1300°F. Hastelloy N

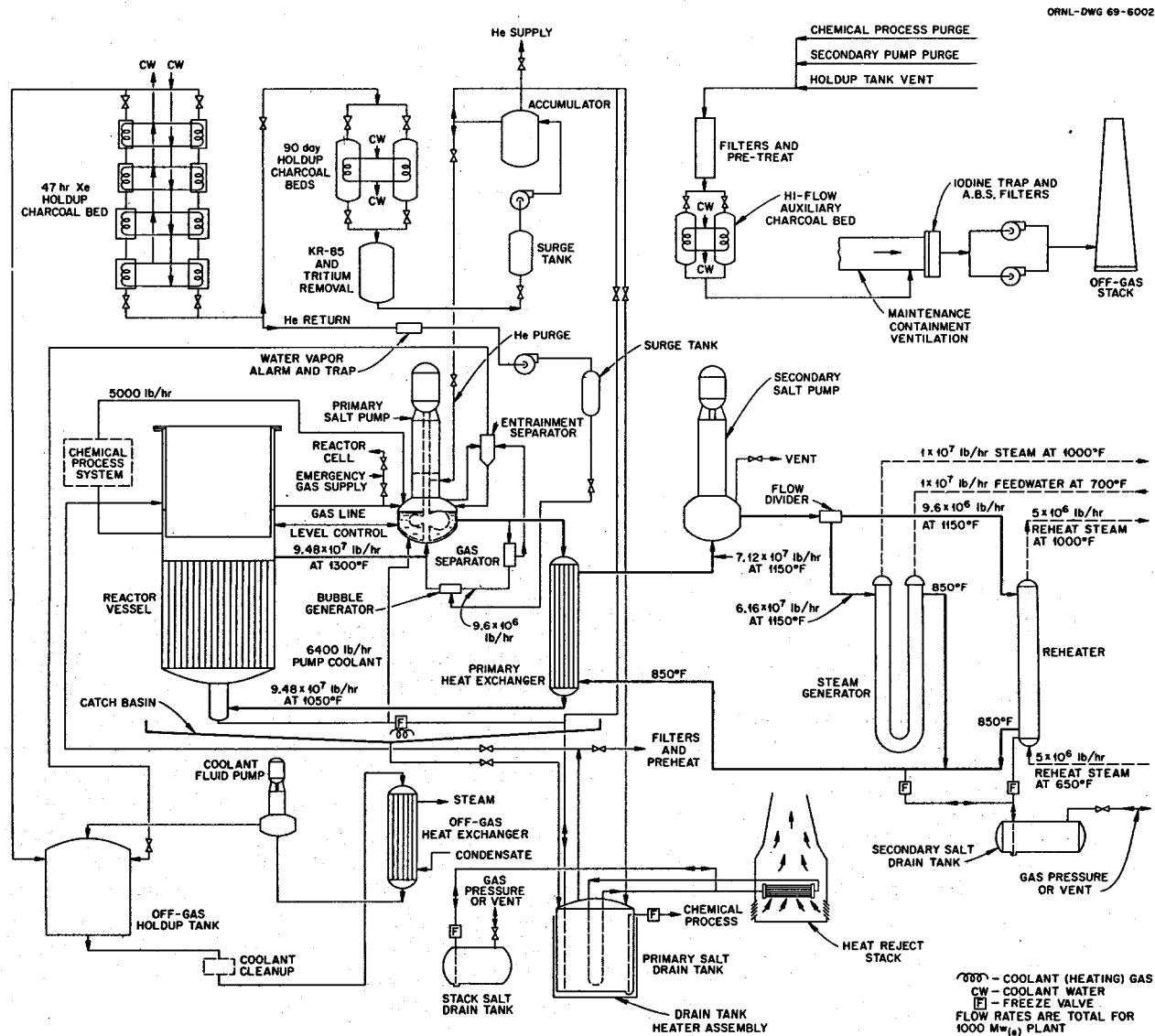


Fig. 5.3. Flow Diagram of 1000-Mw (Electrical) Reactor Plant.

7. Systems and Components Development

Dunlap Scott

7.1 OFF-GAS SYSTEM

A. N. Smith R. B. Gallaher Dunlap Scott

The general flow diagram for the off-gas system is shown in Fig. 5.3 and is still essentially as has been described previously.¹ Current estimates indicate that the gas flow from the reactor primary system to the volume holdup in the off-gas system will be about 11 scfm, of which 9 scfm will come from the gas separators and the balance from the purge streams for the pump shafts and miscellaneous instruments. A liquid-metal coolant operated at temperatures below 500°F is being considered for removing heat from the volume holdup tank. If this method is accepted, the system for cooling the liquid metal would have to be set up separately from the primary-salt drain tank coolant system, since the latter would require a much higher minimum operating temperature. The preliminary design concept for the volume holdup tank is described below.

7.1.1 Volume Holdup Tank

The function of the volume holdup tank is to delay the off-gas stream for about 1 hr to permit decay of a large fraction of the short-lived fission products and dissipation of the accompanying heat. The cooling system must be capable of handling a maximum of 18 Mw of decay heat. Provision must be made for the collection and disposal of 1 to 2 lb of solid fission products per day. Although the incoming gas will be 99+% helium, it will contain trace quantities of a variety of other elements, including: tritium (^3H); stable Kr and Xe; radioactive Kr and Xe and associated daughters such as Ba, Cs, La, Rb, Sr, Y, and Zr; and a "smoke" of the so-called noble metals, Mo, Ru, Tc, and Te. In addition, there will probably be small amounts of the fuel salt in the form of a mist. Such mist particles

will be microspheres and will contain all the ingredients of the fuel stream. In the volume holdup region the noble metals and the daughters of Kr and Xe will be in either the liquid or the solid state, and particles suspended in the gas will probably range in size from about 20 μ down to single atoms.

For the initial approach to the problem, we have assumed that settling or impaction will be used to transfer the particles directly into a coolant stream. The coolant will be recirculated through external equipment for continuous removal of the solids and the associated decay heat. The primary task is the selection of a coolant which will satisfy our tentative requirements for the system. Such a coolant must safely accept and hold the "noble" metals, the daughters of the noble metals, and the daughters of the noble gases and then must release these fission products by treatment only in the proper areas. It must have a vapor pressure low enough to minimize carry-over to the 47-hr holdup system, must be compatible with its container and with fuel salt, and should have an existing technology for use as a heat transfer agent. The eutectic of lead and bismuth appears to offer sufficient attractiveness to merit first place on the list of candidates. It is compatible with the fuel salt, it has high density which should minimize settling out of entrained noble metals, and its low melting point (275°F) may permit operation at low temperature and the use of relatively low-cost alloys as materials of construction.

Figure 7.1 shows a conceptual design of a volume holdup tank utilizing the impaction mechanism for transfer of the solids to a recirculating coolant stream. Performance of the device has been calculated by use of an equation developed for a low-pressure cascade impactor.² The results indicate that it should be possible to design a system of reasonable geometry for

¹MSR Program Semiann. Progr. Rept. Aug. 31, 1968, ORNL-4344, pp. 56 ff. and 72 ff.

²G. W. Parker and H. Buchholz, *Size Classification of Submicron Particles by a Low Pressure Cascade Impactor*, ORNL-4226 (June 1968).

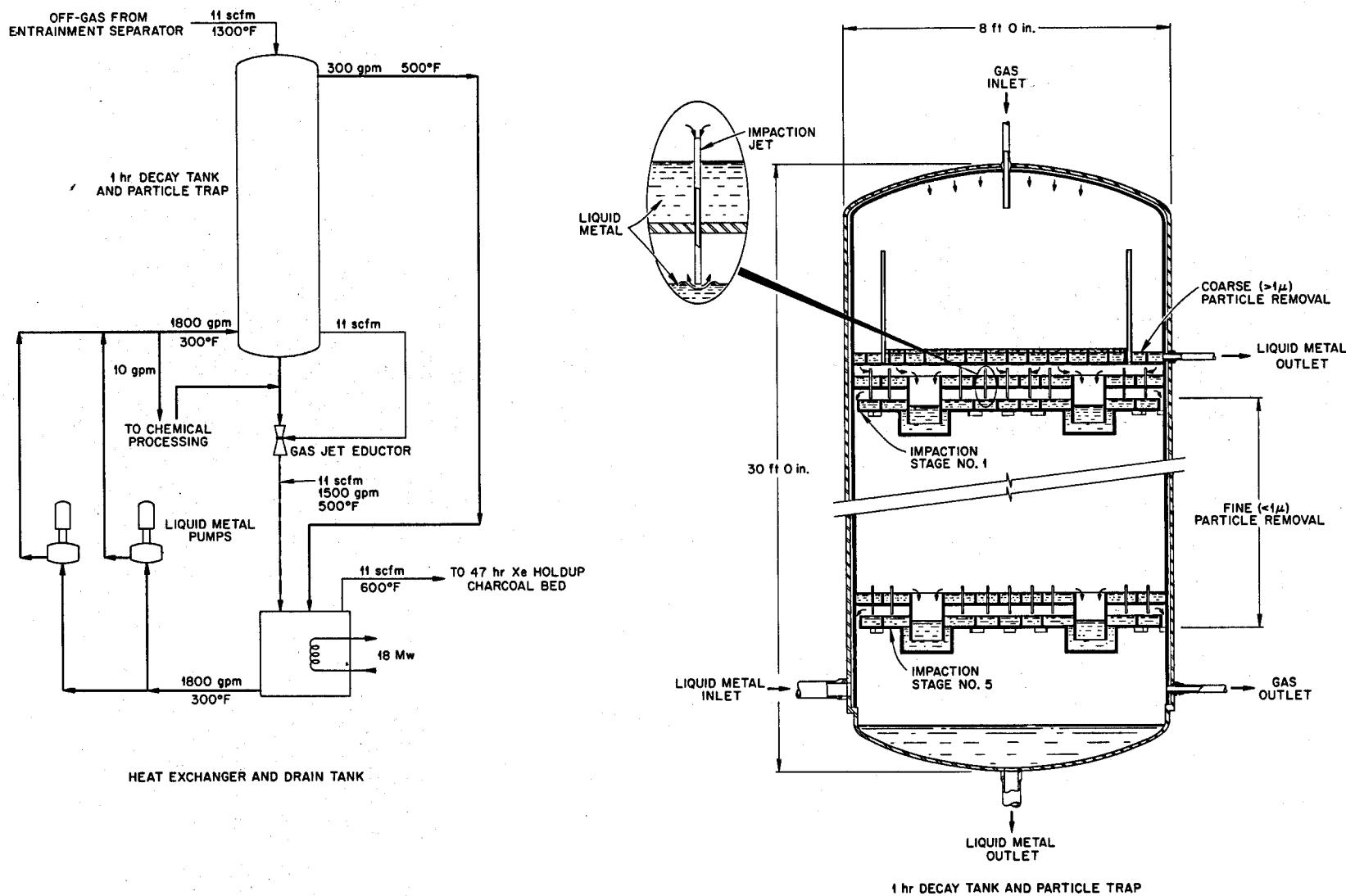


Fig. 7.1. Particle Trap and 1-hr Decay System for MSBR Off-Gas.

removal of particles in the 0.1 -to- $1.0\text{-}\mu$ range. Thus, for removal of $0.15\text{-}\mu$ particles at 50% efficiency, forty-two 0.080-in.-diam jets would be required for one stage to handle the gas flow. A number of stages of relatively low efficiency and operated in series will be used to distribute the heat load and to minimize short-circuiting of the gas into the charcoal beds, which are next in the gas system. The volume holdup tank proper will be 8 ft in diameter by 30 ft high. Coolant flow is from top to bottom, and the coolant is also circulated through a jacket to provide cooling for the walls of the vessel and to absorb the gamma energy which is released as part of the fission product decay.

7.2 NOBLE-GAS MIGRATION IN THE MSBR

R. J. Kedl

In the last semiannual progress report,³ results of ^{135}Xe poisoning calculations were presented for a 1000 Mw (electrical) single-fluid MSBR having a blanket and reflector. The xenon stripping mechanism uses recirculating helium bubbles 0.020 in. in diameter, where the clean helium bubbles are added to the fuel and ^{135}Xe -rich bubbles are stripped out of the fuel in a bypass around the pump. In the calculational model the fuel loop is treated as a well-stripped pot. These computations were repeated for all other noble-gas fission products, of which there are over 30 kryptons and xenons. The results are shown in Table 7.1. Some of the more significant gas-migration parameters are listed below, and the cases are characterized in the table by the ^{135}Xe poison fraction. The tabulated fluxes are about the same for any reasonable combination of these parameters that yield the same ^{135}Xe poison fraction.

^{135}Xe poison fraction, %	1.27	0.56
Volume fraction bubbles in fuel salt	0.002	0.006
Fraction bubbles replaced per loop circuit	0.1	0.1
Xenon diffusion coefficient in bulk graphite, ft^2/hr	10^{-5}	10^{-5}
Xenon diffusion coefficient in graphite coating, ft^2/hr	10^{-7}	10^{-7}
Available void fraction in graphite coating, %	1	1
Graphite coating thickness, in.	0.010	0.010
Mass transfer coefficient to bubbles, ft/hr	2.0	2.0
Graphite surface area in core, ft^2	25,500	25,500

³MSR Program Semiann. Progr. Rept. Aug. 31, 1968, ORNL-4344.

The flow diagram that forms the basis for the calculations in Table 7.1 is shown in Fig. 7.2. Note that the flux into the bubbles is not necessarily equal to the flux out of the reactor. Some longer-lived noble gases are recycled from the 48-hr holdup system back into the reactor with the helium, and for those gases the flux to the bubbles is less than the flux out of the reactor. For the short-lived gases the flux to the bubbles is greater, because some decay and burnup take place between the time the noble gas enters a bubble and the time the bubble leaves the salt. For very short-lived isotopes, where the half-life is a small part of the loop circuit time, one would question the applicability of the well-stirred pot model. Nevertheless, these computations are adequate for the preliminary designs.

7.2.1 Contribution of the Noble Gases to Afterheat in the Graphite

With the use of the noble-gas flux into graphite, described above, we can compute their contribution to

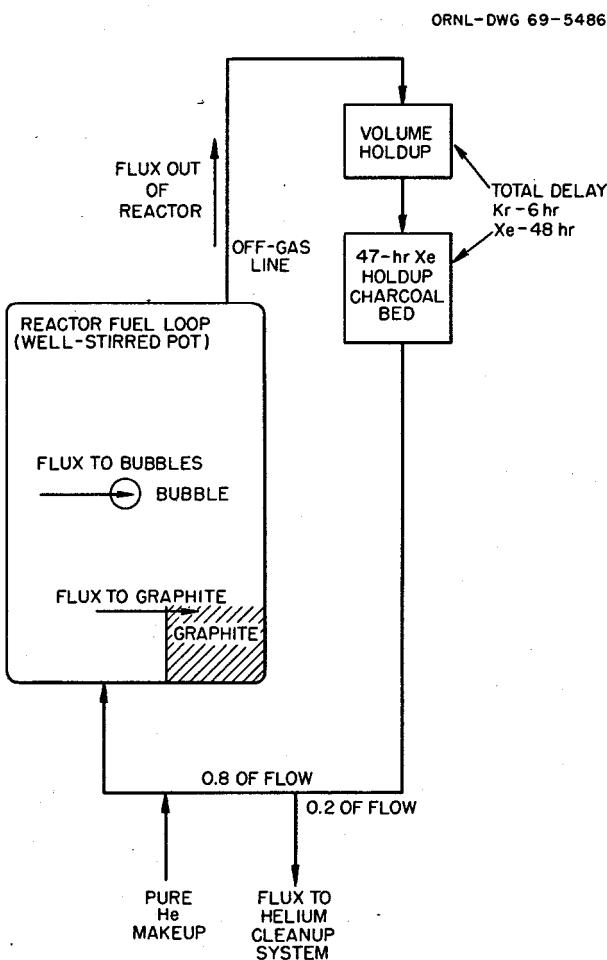


Fig. 7.2. Flow Diagram to Define Terms Used in Table 7.1.

Table 7.1. Noble-Gas Migration in the MSBR

Noble-Gas Isotope	Decay Constant Used in Calculation (hr ⁻¹)	Cumulative Yield from ²³³ U (fraction)	Thermal Cross Section (barns)	1 ³⁵ Xe Poison Fraction = 1.27%				1 ³⁵ Xe Poison Fraction = 0.56%			
				Noble-Gas Flux to Graphite (atoms/hr)	Noble-Gas Flux to Bubbles (atoms/hr)	Noble-Gas Flux Out of Reactor (atoms/hr)	Noble-Gas Flux to Helium Cleanup System (atoms/hr)	Noble-Gas Flux to Graphite (atoms/hr)	Noble-Gas Flux to Bubbles (atoms/hr)	Noble-Gas Flux Out of Reactor (atoms/hr)	Noble-Gas Flux to Helium Cleanup System (atoms/hr)
⁸² Kr	1.0×10^{-5}	0.003	45.0	3.71×10^{18}	2.10×10^{21}	1.05×10^{22}	2.10×10^{21}	1.24×10^{18}	2.11×10^{21}	1.05×10^{22}	2.11×10^{21}
⁸³ Kr	1.0×10^{-5}	0.0114	0	8.56×10^{17}	3.00×10^{21}	1.50×10^{22}	3.00×10^{21}	2.85×10^{17}	3.00×10^{21}	1.50×10^{22}	3.00×10^{21}
⁸⁴ Kr	1.0×10^{-5}	0.0110	0.160	8.42×10^{17}	2.90×10^{21}	1.45×10^{22}	2.90×10^{21}	2.81×10^{17}	2.90×10^{21}	1.45×10^{22}	2.90×10^{21}
⁸⁵ Kr	7.35×10^{-6}	0.0249	0.096	1.40×10^{18}	6.56×10^{21}	3.28×10^{22}	6.56×10^{21}	4.67×10^{17}	6.56×10^{21}	3.28×10^{22}	6.56×10^{21}
⁸⁶ Kr	1.0×10^{-5}	0.0328	0.060	2.48×10^{18}	8.64×10^{21}	4.32×10^{22}	8.64×10^{21}	8.27×10^{17}	8.64×10^{21}	4.32×10^{22}	8.64×10^{21}
⁸⁷ Kr	0.547	0.0450	500.0	1.87×10^{21}	9.55×10^{21}	9.67×10^{21}	7.27×10^{19}	7.15×10^{19}	1.10×10^{22}	1.11×10^{22}	8.35×10^{19}
⁸⁸ Kr	0.247	0.0570	0	2.34×10^{21}	1.24×10^{21}	1.50×10^{22}	6.87×10^{20}	8.81×10^{20}	1.40×10^{22}	1.70×10^{22}	7.72×10^{20}
⁸⁹ Kr	13.0	0.0623	0	1.77×10^{21}	7.08×10^{21}	5.07×10^{21}	0	9.49×10^{20}	1.14×10^{22}	8.16×10^{21}	0
⁹⁰ Kr	75.6	0.0555	0	7.22×10^{20}	2.01×10^{21}	6.08×10^{20}	0	5.66×10^{20}	4.74×10^{21}	1.43×10^{21}	0
⁹¹ Kr	249.0	0.0410	0	2.51×10^{20}	5.28×10^{20}	6.13×10^{19}	0	2.28×10^{20}	1.44×10^{21}	1.67×10^{20}	0
⁹² Kr	832.0	0.0296	0	7.17×10^{19}	1.21×10^{20}	4.58×10^{18}	0	6.96×10^{19}	3.53×10^{20}	1.33×10^{19}	0
⁹³ Kr	1230.0	0.0142	0	2.48×10^{19}	3.97×10^{19}	1.03×10^{18}	0	2.43×10^{19}	1.17×10^{20}	3.02×10^{18}	0
⁹⁴ Kr	2496.0	0.0062	0	5.88×10^{18}	8.65×10^{18}	1.12×10^{17}	0	5.82×10^{18}	2.57×10^{19}	3.32×10^{17}	0
⁹⁵ Kr	2490.0	0.0019	0	1.80×10^{18}	2.65×10^{18}	3.43×10^{16}	0	1.78×10^{18}	7.87×10^{18}	1.02×10^{17}	0
¹²⁶ Xe	1.0×10^{-5}	0.0020	1.50	3.90×10^{17}	5.27×10^{20}	2.63×10^{21}	5.25×10^{17}	1.30×10^{17}	5.27×10^{20}	2.63×10^{21}	5.26×10^{20}
¹²⁸ Xe	1.0×10^{-5}	0.0002	2.50	4.08×10^{16}	5.00×10^{19}	2.50×10^{20}	4.99×10^{19}	1.36×10^{16}	5.00×10^{19}	2.50×10^{20}	4.99×10^{19}
¹²⁹ Xe	1.0×10^{-5}	0.0210	0	3.48×10^{18}	5.53×10^{21}	2.76×10^{22}	5.52×10^{21}	1.16×10^{18}	5.53×10^{21}	2.76×10^{22}	5.52×10^{21}
¹³⁰ Xe	1.0×10^{-5}	0.0010	2.50	2.21×10^{17}	2.71×10^{20}	1.35×10^{21}	2.71×10^{20}	7.38×10^{20}	2.71×10^{20}	1.35×10^{21}	2.71×10^{20}
¹³¹ Xe	1.0×10^{-5}	0.0385	120.0	9.42×10^{19}	1.00×10^{22}	5.02×10^{22}	1.00×10^{22}	3.16×10^{19}	1.01×10^{22}	5.05×10^{22}	1.01×10^{22}
¹³² Xe	1.0×10^{-5}	0.0548	0.20	9.30×10^{18}	1.44×10^{22}	7.20×10^{22}	1.44×10^{22}	3.10×10^{18}	1.44×10^{22}	7.20×10^{22}	1.44×10^{22}
¹³³ Xe	5.48×10^{-3}	0.0648	190.0	2.36×10^{21}	1.47×10^{22}	3.82×10^{22}	5.87×10^{21}	8.68×10^{20}	1.62×10^{22}	4.21×10^{22}	6.47×10^{21}
¹³⁴ Xe	1.0×10^{-5}	0.0683	0.20	1.16×10^{19}	1.80×10^{22}	8.97×10^{22}	1.79×10^{22}	3.87×10^{18}	1.80×10^{22}	8.98×10^{22}	1.79×10^{22}
¹³⁵ Xe	0.0753	0.0616	1.05×10^6	4.61×10^{21}	1.12×10^{22}	1.13×10^{22}	6.11×10^{19}	1.93×10^{21}	1.41×10^{22}	1.43×10^{22}	7.68×10^{19}
¹³⁶ Xe	1.0×10^{-5}	0.0700	0.15	1.18×10^{19}	1.84×10^{22}	9.20×10^{22}	1.84×10^{22}	3.94×10^{18}	1.84×10^{22}	9.20×10^{22}	1.84×10^{22}
¹³⁷ Xe	9.90	0.0716	0	3.87×10^{21}	8.49×10^{21}	6.51×10^{21}	0	2.04×10^{21}	1.34×10^{22}	1.03×10^{22}	0
¹³⁸ Xe	2.446	0.0663	0	4.53×10^{21}	1.09×10^{22}	1.01×10^{22}	0	2.02×10^{21}	1.45×10^{22}	1.35×10^{22}	0
¹³⁹ Xe	60.85	0.0493	0	1.19×10^{21}	2.09×10^{21}	7.31×10^{20}	0	9.01×10^{20}	4.75×10^{21}	1.66×10^{21}	0
¹⁴⁰ Xe	156.0	0.0352	0	4.45×10^{20}	6.90×10^{20}	1.19×10^{20}	0	3.87×10^{20}	1.80×10^{21}	3.12×10^{20}	0
¹⁴¹ Xe	1250.0	0.0180	0	3.80×10^{19}	4.96×10^{19}	1.26×10^{18}	0	3.73×10^{19}	1.46×10^{20}	3.71×10^{18}	0
¹⁴² Xe	1660.0	0.0163	0	2.64×10^{19}	3.40×10^{19}	6.55×10^{17}	0	2.60×10^{19}	1.00×10^{20}	1.94×10^{18}	0
¹⁴³ Xe	2490.0	0.0017	0	1.86×10^{18}	2.34×10^{18}	3.04×10^{16}	0	1.84×10^{18}	6.96×10^{18}	9.01×10^{16}	0
¹⁴⁴ Xe	2490.0	0.0001	0	7.74×10^{16}	9.77×10^{16}	1.27×10^{15}	0	7.66×10^{16}	2.90×10^{17}	3.76×10^{15}	0

afterheat. Figure 7.3 shows the results for the case where ^{135}Xe poison fraction is 0.56%. It is assumed that the noble-gas flux into the graphite is constant and continuous for two years with the reactor at power. The amount of noble gases and their daughters can then be computed after this period of time. This computation is based on straight-chain decay and no side decay loops.

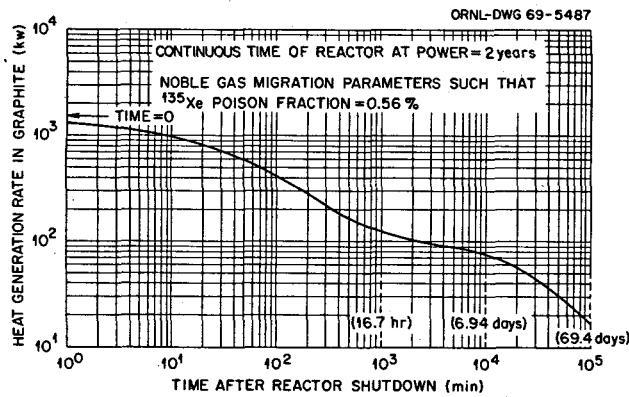


Fig. 7.3. Afterheat Contribution by Noble Gases and Their Daughters Adsorbed by the Graphite in the MSBR Core [1000 Mw (Electrical)]. The residence time for krypton in the charcoal beds is about one-twelfth of that for xenon. The flux of atoms into the off-gas line was calculated by R. J. Kedl using 0.56% poison fraction. Decay heat calculations were made by M. Bell using computer code.

7.2.2 Distribution of Decay Heat in the MSBR Off-Gas System

A. N. Smith M. Bell

An estimate has been made of the distribution of fission product decay heat in the 1000 Mw (electrical) MSBR off-gas system. The calculations were based on the following model:

1. The flux of krypton and xenon into the off-gas line will be as calculated by R. J. Kedl for 0.56% poison fraction (see Table 7.1). Solid daughters of krypton and xenon are assumed to plate out at the point of formation, which gives a high heating value for the pipe sections.

2. The flux to the off-gas will include 50% of the noble-metal fission products. Of these, 100% will be retained in the 1-hr volume holdup.

3. Krypton delay in the charcoal beds will be $\frac{1}{12}$ xenon delay.

4. The off-gas system will be divided into 20 regions, as shown in Fig. 7.4. The unstable noble gases will decay exponentially in accordance with an assigned delay or residence time. The 1-hr volume holdup and the 47-hr xenon delay charcoal bed were divided into compartments with various delay times in an attempt to obtain approximately equal heat loads. The delay times for the pipe sections were arbitrarily set at 18 sec each.

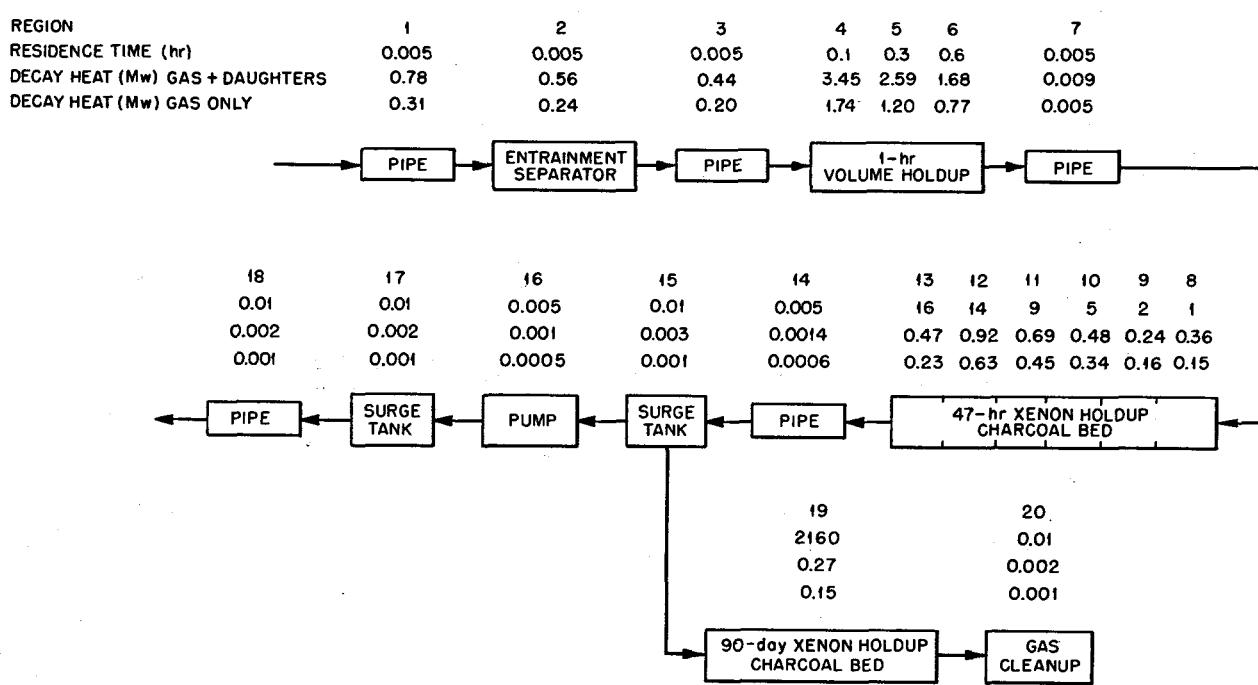


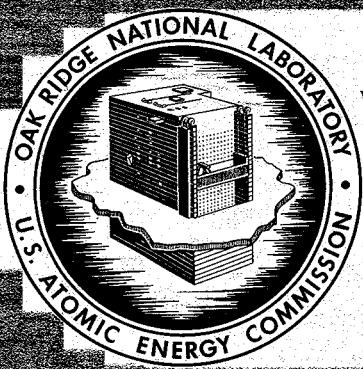
Fig. 7.4. Distribution of Decay Heat in MSBR Off-Gas System.

ORNL-4344
UC-80 - Reactor Technology

MOLTEN SALT REACTOR PROGRAM

SEMIANNUAL PROGRESS REPORT

FOR PERIOD ENDING AUGUST 31, 1968



OAK RIDGE NATIONAL LABORATORY

operated by

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U. S. ATOMIC ENERGY COMMISSION

7. Systems and Components Development

Dunlap Scott

7.1 NOBLE-GAS MIGRATION IN THE MSBR REFERENCE DESIGN

R. J. Kedl

Noble-gas migration in a conceptual version of the two-fluid MSBR was discussed in some detail in a previous progress report.¹ The concept of using circulating helium bubbles for stripping noble gases from the fuel salt was presented. At that time, all the bubbles were injected into the fuel salt at the core outlet and removed from the salt at the core inlet. The objective was to keep the core nominally free of bubbles and to avoid any effects that they might have on reactivity. A recent change in ground rules allows up to 1% bubbles by volume of salt in the core. This greatly simplified the noble-gas stripping problem, because we can now recirculate the bubbles many times around the fuel loop and let them approach much closer to saturation. The result is that the volumetric flow rate of helium in the gas system is considerably reduced. Also, the bubble-generating and -removal equipment may be put into a side stream rather than in the main line of the fuel loop.

Xenon-135 poisoning calculations have been made for the single-fluid MSBR reference design, described in a preceding section, and include the recirculating bubble concept described above. The analytical model used is that of a well-stirred pot; that is, the concentration of xenon dissolved in salt and in the bubbles is taken to be constant around the entire fuel loop. Figure 7.1 shows the effect of the amount of helium bubble recirculation on ^{135}Xe poison fraction. The other variable parameter on this plot is the average void percent of bubbles in

the fuel loop, which is almost exactly equal to the average void percent in the core. Bubble surface areas are computed for bubbles 0.020 in. in diameter. Other constant parameters are as listed in the figure legend. The graphite is taken to be uncoated

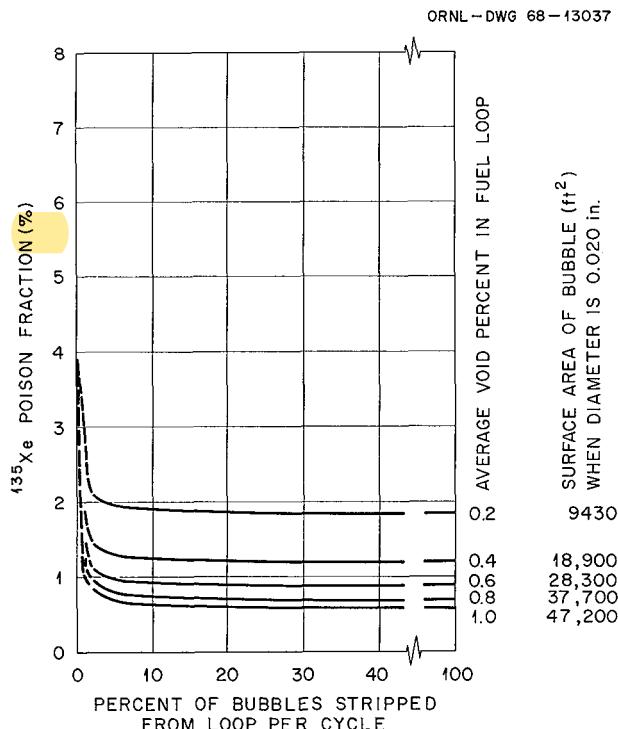


Fig. 7.1. ^{135}Xe Poison Fraction in Reference Design MSBR as a Function of the Percent Bubbles Stripped from Fuel Loop per Cycle. Parameters: reactor power, 1000 Mw (electrical); diffusion coefficient of Xe in graphite at 1200°F , $10^{-5} \text{ ft}^2/\text{hr}$; graphite void fraction available to Xe, 10%; total yield of ^{135}Xe , 6.16%; ^{135}Xe thermal cross section, 1.049×10^6 barns; mass transfer coefficient to bubbles, 2.0 ft/hr .

¹MSR Program Semiann. Progr. Rept. Feb. 29, 1968, ORNL-4254, p. 74.

and has a diffusion coefficient for xenon of 10^{-5} ft²/hr at 1200°F (permeability for helium at room temperature $\sim 10^{-5}$ cm²/sec). From this plot, it can be seen that the fraction of bubbles stripped per loop cycle can be 10% or less before the back pressure of xenon in the bubble starts to significantly reduce the stripping efficiency. This means that for a given void fraction in the fuel salt, the bubbles can be recirculated ten times before they must be removed, and the capacity of the gas system will be $\frac{1}{10}$ of that if all bubbles are replaced per cycle. As pointed out in the previous semiannual report, the mass transfer coefficient to circulating bubbles is one of the least-known parameters in this calculation.

A literature survey and analysis has been made² on the expected mass transfer coefficient to the bubbles in a turbulent stream. For the rigid interface model the expected value will be about 2.0 ft/hr and for the mobile interface model the expected value will be about 13.0 ft/hr. The above plot was computed using a value of 2.0 ft/hr. It is unlikely that a completely mobile interface will

exist with these very small bubbles, but a partially mobile interface may be realized. Now in these calculations, the variables "bubble surface area" and "bubble mass transfer coefficient" always appear as a product. Figure 7.1 can then be used to estimate the effect of a change in the mass transfer coefficient. In this plot the average void percent bubbles in the fuel is used only to generate a surface area of 20-mil-diam bubbles and otherwise is a negligible parameter if the bubbles are far from saturation. The bubbles do not approach saturation until the fraction of bubbles stripped from the loop per cycle is less than 5%. Therefore, in the straight-line section of these curves, doubling the surface area is the same as doubling the mass transfer coefficient. For example, if the mass transfer coefficient is doubled for the case where the bubble surface area is 9430 ft², the poison fraction would be reduced from 1.9% to 1.25% for the 10% bubble stripping fraction. It should also be noted that for a given void fraction, the bubble surface area is reduced by a factor of 2 if the bubble diameter is doubled.

Figure 7.2 shows the effect of a very low permeability coating on the bulk graphite. This is envisioned as a sealant of pyrolytic graphite, or possibly metal, that plugs the surface pores of the bulk

²F. N. Peebles, Removal of Xenon-135 from Circulating Fuel Salt of the MSBR by Mass Transfer to Helium Bubbles, ORNL-TM-2245 (July 23, 1968).

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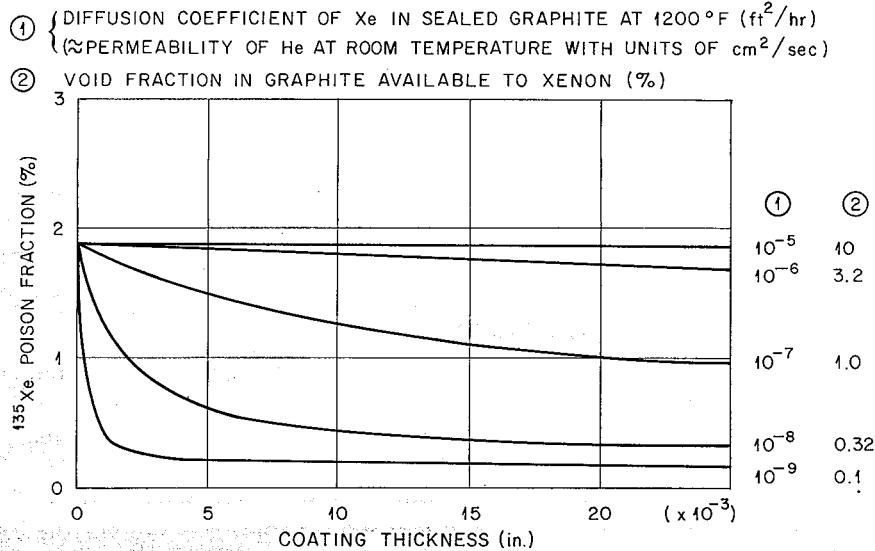


Fig. 7.2. 135Xe Poison Fraction in Reference Design MSBR as a Function of Graphite Sealing Parameters.

Parameters: reactor power, 1000 Mw (electrical); diffusion coefficient of Xe in bulk graphite at 1200°F, 10^{-5} ft²/hr; bulk graphite void fraction available to Xe, 10%; average void percent of bubbles in fuel loop, 0.2%; bubble surface area (0.020 in. diameter), 9430 ft²; percent bubbles stripped per loop cycle, 10%.

graphite. To develop this illustration, parameters were chosen from Fig. 7.1 to yield a high poison fraction ($\sim 1.9\%$). With these parameters the calculations were repeated to obtain the effects of the permeability and thickness of the sealed layer on the poison fraction. In this calculation it was assumed that the void fraction in graphite available to xenon decreased by one order of magnitude when the permeability decreased by two orders of magnitude. It can be seen that permeabilities of 10^{-7} and less are quite good in reducing the poison fraction.

The target ^{135}Xe poison fraction in the MSBR is 0.5% or less. With circulating bubbles alone (Fig. 7.1), this may or may not be attainable. Some of the uncertainties are the bubble mass transfer coefficient, the ability to generate uniformly 0.020-in. bubbles, and the ability of the bubbles to maintain their identity for many circuits around the fuel loop (e.g., not agglomerate or migrate to a surface). If, however, the circulating bubbles are assisted by sealed graphite, the target poison fraction can be achieved and surpassed.

7.2 BUBBLE GENERATOR

R. J. Kedl

Circulating helium bubbles with the fuel salt is the preferred method of stripping noble gases generated by fission in the MSBR. Xenon-135 poisoning calculations reported above are based on surface areas generated by bubbles 0.020 in. in diameter. We have conducted exploratory tests with water and air for some time to try to generate bubbles of this diameter with equipment that can be scaled up to sizes adequate for the MSBR. In general, two avenues of approach have been taken: (1) mechanically powered generators, where a mixture of air and water is fed to a mechanically agitating geometry that breaks up the large bubbles into very small ones; and (2) fluid-powered generators, where, for example, air is injected into the throat of a venturi. Without going into details of the several models tested, the following conclusions have been reached.

1. Emphasis on mechanically powered generators should be reduced. During the early development work, smaller and more uniformly sized bubbles were attained by this method than with fluid-powered generators; however, other potential problems would be considerable. For instance, a me-

chanical drive motor would be needed, but a separate drive motor would be undesirable. The most logical place to install a mechanical generator would be on the impeller of the pump. This complicates an already critical item; furthermore, estimates indicate that the most successful generator model tested would cavitate in MSBR-scaled equipment. A great deal of development work would be needed to successfully attach a mechanical bubble generator to a pump impeller.

2. Emphasis on the fluid-powered generator should be increased. The most successful fluid-powered bubble generator we have run is a venturi modified so that gas is injected into small holes in the throat and turbulence in the diffuser region generates the bubbles. Quantitative information is not yet available; however, the bubbles appear to be larger than desired — in the order of 50 mils. It is hoped that further experimentation will reduce this. The venturi has the additional advantage that it could serve as the helium pump if it were placed on the suction side of the fuel pump.

7.3 BUBBLE SEPARATOR

R. J. Kedl

Work is under way on the development of a pipeline bubble separator for the removal of noble-gas-rich bubbles from the fuel salt. The separator is a straight section of 4-in. pipe about 4 ft long with swirl vanes at the inlet end and recovery vanes at the outlet. The swirling fluid generates a high gravitational field in the pipe, and circulating bubbles will migrate to a gas-filled vortex in the center of the pipe. The recovery vanes straighten out the fluid and recover some of its energy. The gas take-off is in the hub of the recovery vanes.

A full-scale model has been built and is operating in an existing water loop. A photograph of the separator operating in a Plexiglas pipe is shown in Fig. 7.3. Two sets of swirl vanes have been tested. The second set operates better than the first, and the results of this set only will be reported. Velocity profiles were measured in a radial direction and are shown in Fig. 7.4. Axial and tangential components are very nearly the same and are shown as a single line. It was also noted that there was very little attenuation of the velocity over the $2\frac{1}{2}$ -ft length between the vanes. At 660 gpm the following pressure distributions were observed:

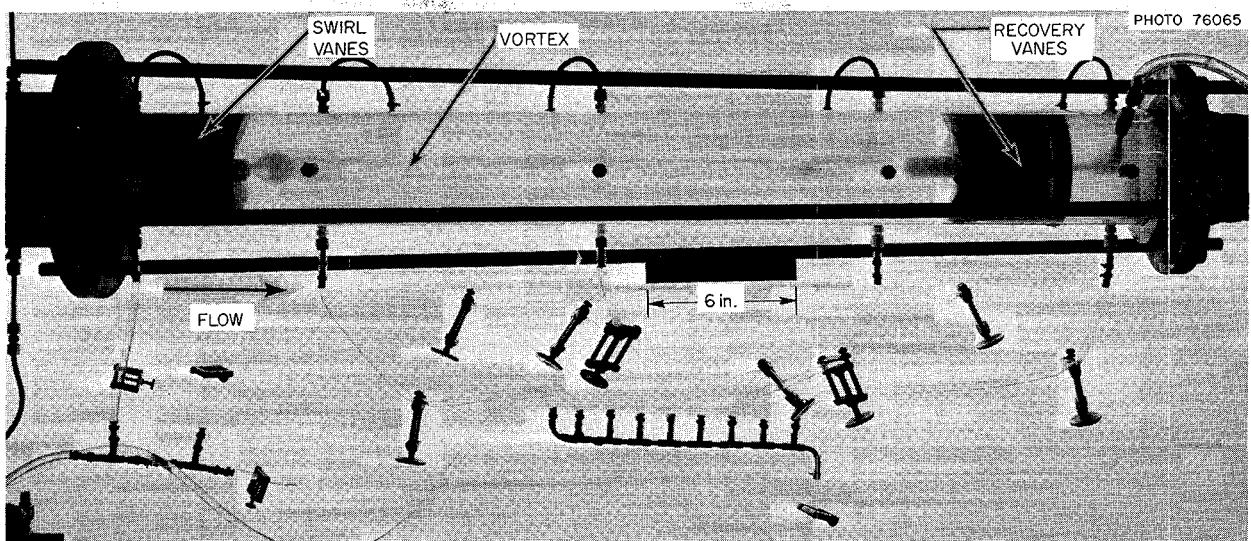


Fig. 7.3. Bubble Separator Model.

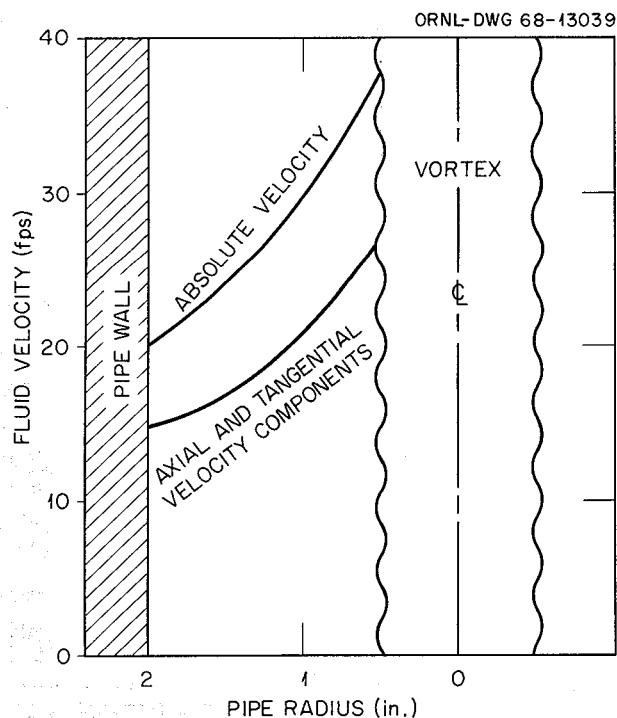


Fig. 7.4. Radial Velocity Distributions in Pipeline Bubble Separator.

Pressure drop across swirl vanes at wall 3.7 ft of fluid

Radial pressure difference in swirling fluid from wall to vortex 26 ft

Net pressure loss across recovery vanes at wall 5.1 ft

The only problem encountered is that the vortex does not attach to the hub of the recovery vanes. It can be seen in the photograph that the vortex seems to disintegrate just ahead of the recovery vanes, and any bubbles entering this area are swept on past the hub. In earlier work done on pipeline gas separators³ this phenomenon had not been experienced. The trouble is probably due to the hub size. It is planned to make a new set of vanes with the hub diameter doubled, which will be more in accordance with the separator of the earlier work.

7.4 SODIUM FLUOROBORATE CIRCULATING TEST LOOP

A. N. Smith P. G. Smith

Circulation of the flushing charge of salt was started on March 4, 1968, and was continued in a

³J. A. Hafford, *Development of the Pipeline Gas Separator*, ORNL-1602 (February 1954).

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always less than 0.01 of the combined tritium concentration, the presence of any elemental tritium does not significantly affect this observation.

The concentration of elemental tritium in the off-gas samples rose during the addition phase of each experiment and apparently began to decrease as soon as the addition was stopped. The maximum concentration in the first test was about 800 pCi/cm³ and only 40 pCi/cm³ in the second. In both cases the decrease in concentration with time after the addition was too irregular to justify any quantitative evaluation.

Although no measure of elemental tritium concentration in the salt is available, a value can be inferred from the concentration in the off-gas by (1) assuming that the elemental tritium in the off-gas samples represents release from the salt and only from the salt, and (2) assigning reasonable values to gas stripping parameters in the CSTF pump tank. Concentrations of elemental tritium calculated in this way indicate that the ratios of combined/elemental tritium in the salt were about 50 and 530 in the first and second experiments respectively. It appears that chemical interactions between the tritium-containing compound in the off-gas and the new metal of the sample line may have been responsible for the high concentrations of elemental tritium in the off-gas samples from the first test and that the actual ratio of combined/elemental tritium in the salt may have been higher than 50.

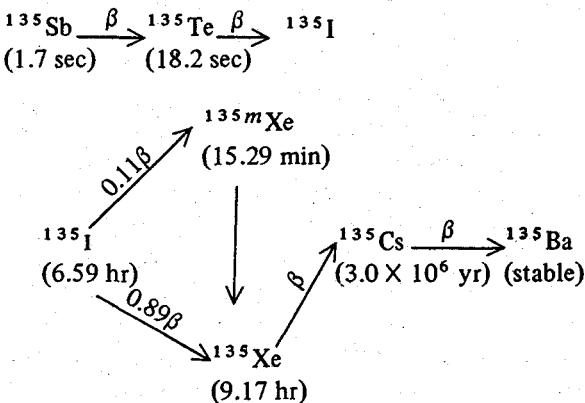
The inferred maximum concentration of elemental tritium in the salt during the second experiment is about 0.13 nCi/g. Extension of the calculated tritium distribution with nominal metal-wall permeability (Table 1.2) to lower concentrations indicates that, at 0.13 nCi/g, tritium permeation through the loop walls could account for no more than about one-third of the tritium added to the system. Since this is close to the amount not accounted for in the off-gas samples, it appears that the effective permeability of the loop walls is near that of bare metal.

1.2 XENON BEHAVIOR IN THE MSBR

G. T. Mays

The computer program MSRXP (Molten-Salt Reactor Xenon Poisoning) describing the ¹³⁵Xe behavior in the reference-design MSBR was used to perform calculations to study the effects of the Knudsen diffusion coefficient for xenon in the bulk graphite and graphite coating of the reactor core on the ¹³⁵Xe poison fraction. The program has been described previously.^{6,7}

Following the fission of the fuel, the decay of the mass-135 fission fragments is assumed to follow the decay chain shown below:



This diagram illustrates the half-life of each isotope and the branching ratio of the ¹³⁵I decaying to ^{135m}Xe and ¹³⁵Xe assumed for this study. Along with this decay chain the following input data were used:

Bubble concentration: 44 bubbles per cubic centimeter of salt

Total helium dissolved in salt and present in gas bubbles: 1.0×10^{-6} mole/cm³

Bubble separator efficiency: 90%

For these conditions the mass transfer correlation in the program gives a bubble mass transfer coefficient of 0.0166 cm/sec, which leads to a loop-averaged void fraction of 0.55% with an average bubble diameter of 0.65 mm. The calculated ¹³⁵Xe poison fraction is 0.0046.

The reference Knudsen diffusion coefficients for the bulk graphite and graphite coating associated with the 0.0046 poison fraction are 2.58×10^{-9} and 2.58×10^{-6} cm²/sec respectively.* The bulk graphite values were varied from 2.58×10^{-9} to 2.58×10^{-6} cm²/sec, assuming no graphite coating was present (Table 1.3, cases 1, 3, 5, 7), to observe the effect on the poison fraction. The low-permeability graphite coating - 0.28 mm thick - was assigned bulk-graphite values for the Knudsen diffusion coefficient and porosity, making the coating part of the bulk graphite for calculation purposes. Under these conditions the porosity of the bulk graphite was held constant at a value about 31 times

*The complete units for diffusion coefficient are (cm³ gas)/(sec · cm graphite).

6. H. A. McLain et al., in *MSR Program Semiannual Prog. Rep. Aug. 31, 1972*, ORNL-4832, pp. 11, 13.

7. H. A. McLain et al., in *MSR Program Semiannual Prog. Rep. Feb. 29, 1972*, ORNL-4782, pp. 13, 16-17.

Table 1.3. ^{135}Xe poison fraction as a function of the Knudsen diffusion coefficient for the graphite coating and bulk graphite of the reactor core

Bulk graphite ^a	Knudsen diffusion coefficient (cm ³ gas/sec · cm graphite)	Graphite coating	Calculated ^{135}Xe poison fraction
1.	2.58×10^{-6} ^b	No coating	0.0153
2.	2.58×10^{-6}	2.58×10^{-6}	0.0152
3.	2.58×10^{-7}	No coating	0.0140
4.	2.58×10^{-6}	2.58×10^{-7}	0.0145
5.	2.58×10^{-8}	No coating	0.0113
6.	2.58×10^{-6}	2.58×10^{-8}	0.0107
7.	2.58×10^{-9}	No coating	0.0077
8.	2.58×10^{-6}	2.58×10^{-9} ^c	0.0046 ^d

^aBulk graphite porosity value remains constant in all cases, ~ 31 times greater than the value for the graphite coating.

^bReference value for bulk graphite.

^cReference value for graphite coating.

^dPoison fraction for reference case.

greater than that of the graphite coating.* In addition, the Knudsen diffusion coefficient for the graphite coating was varied within the same, aforementioned range while the diffusion coefficient of the bulk graphite was held constant at its reference value of 2.58×10^{-6} cm²/sec to observe the effects on the poison fraction (cases 2, 4, 6, 8). The previously stated values involving bubble characteristics and mass transfer were held constant throughout this series of calculations.

The results (Table 1.3) indicate that Knudsen diffusion coefficients for the bulk graphite and graphite coating at least as low as the reference values (2.58×10^{-6} and 2.58×10^{-9} cm²/sec, i.e., case 8) would be required to meet the 0.005 target value for the ^{135}Xe poison fraction. A diffusion coefficient of less than 2.58×10^{-9} cm²/sec would be required for the bulk graphite with no coating, because of its higher porosity. If the permeability of the graphite coating did not yield a diffusion coefficient equal to that of the reference value, such a coating would have little effect on xenon poisoning. The penalty for not coating the graphite is about 0.01 in xenon poison fraction, or 0.01 in breeding ratio, if no attempt is made to decrease the permeability or porosity of the base material.

It may be noted in case 3 that a slight reduction in the Knudsen diffusion coefficient for the bulk graphite is

*In practice, substantial reductions in the Knudsen diffusion coefficient probably would be accompanied by reduced porosity.

more effective in reducing the ^{135}Xe poison fraction than a similar reduction in the Knudsen diffusion coefficient for the graphite coating in case 4. In cases 6 and 8, where the permeability of the coating is very low, reducing the Knudsen diffusion coefficient for the graphite coating affects the ^{135}Xe poison fraction much more strongly.

1.3 NEUTRONIC ANALYSIS

H. T. Kerr D. L. Reed E. J. Allen

The neutronic analysis work during this reporting period has involved several tasks aimed at additional description of the neutronic characteristics of an MSBR and the provision of neutronics information for the fueled in-reactor irradiation experiments.

1.3.1 MSBR Studies

Neutronic analysis studies for the reference-design MSBR are in progress in three areas:

1. development of a two-dimensional neutronic computational model of the MSBR, using the computer code VENTURE and reestablishing the operability of a reactor optimization code (ROD),
2. updating the neutron cross-section data base used by various computer programs,
3. calculation of the rate at which helium will be produced in the reactor vessel of the MSBR.

Computational models. A neutronic computational model of the MSBR, using the computer code VENTURE,⁸ is being developed. VENTURE is a multi-dimensional, multigroup, neutron diffusion computer code. The MSBR model will have nine neutron energy groups and the (r-z) geometry shown in Fig. 1.3. The various zones in the model allow for different core compositions and cross-section sets. In addition to providing a check of the design studies made with the ROD⁹ code using one-dimensional calculations, this model will permit explicit evaluation of the nuclear reactivity effects associated with localized core perturbations, such as limited core voiding. Previously such effects were conservatively estimated from calculations for an infinite medium of salt and graphite.

8. T. B. Fowler, D. R. Vondy, and G. W. Cunningham III, *VENTURE: A Code Block for Solving Multigroup Neutron Problems Applying the Finite-Difference Diffusion-Theory Approximation to Neutron Transport*, ORNL-5062 (October 1975).

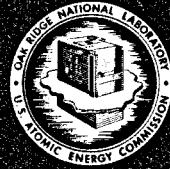
9. H. F. Baumann et al., *ROD: A Nuclear and Fuel Cycle Analysis Code for Circulating-Fuel Reactors*, ORNL-TM-3359 (September 1971).

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putations was made to more carefully evaluate fission-product poisoning so as to determine the validity of the ROD results. The approach chosen was to adopt a fission-product treatment⁶ originally developed for solid-fueled reactors and to adjust the yields to allow for the chemical behavior of the nuclides in a circulating-fuel system. The basic treatment deals explicitly with 26 nuclides in 11 chains and then adds two pseudoelements to account for lumped long-term effects of slowly saturating and nonsaturating fission products. The actual treatment used was reduced by several nuclides to allow (as in ROD) for the rapid disappearance of noble-metal and noble-gas fission products from the circulating system. The daughters of noble-gas radionuclides having half-lives longer than 40 sec were assumed to be removed from the reactor fuel salt, whereas daughter-products of noble-gases having shorter half-lives were assumed to be produced in the salt at their full fission yield.

The resultant fission-product set was described for treatment by the CITATION code,⁷ which evaluates, explicitly, the concentration of each nuclide as a function of power history. Input data required for the CITATION treatment include fission yields (as functions of fissile nuclides), decay constants, and energy-dependent cross sections. The fission yields and decay constants for those nuclides treated explicitly were taken from the tabulation by Bell,⁸ whereas those for the pseudoelements were based on Bennett's values.⁶ The cross-section data were obtained from the 123-energy-group G123 library in XSDRN.⁹ These were processed with XSDRN to produce composition-, energy-, and geometry-weighted cross sections in 87 energy groups for the particular reactor configuration. The final cross-section set included 30 energy groups in the thermal range $E < 1.86$ eV to ensure reasonable accounting for the large resonance peak in the ^{240}Pu cross section at 1 eV. The CITATION calculations were then done in 87 energy groups with a point model of the reactor to evaluate the fission-product poison fraction as a function of time.

6. L. L. Bennett, *Recommended Fission-Product Chains for Use in Reactor Evaluation Studies*, ORNL-TM-1658 (Sept. 1966).

7. T. B. Fowler, D. R. Vondy, and G. W. Cunningham, *Nuclear Reactor Core Analysis Code: CITATION*, ORNL-TM-2496, Rev. 2 (July 1971).

8. M. J. Bell, *Computer Code for the Solution of Large Systems of Simultaneous Linear Equations: Application to a 2200 MW(th) Single Region Molten-Salt Reactor*, ORNL-CF-687-32 (July 1968).

9. C. W. Craven, Jr., and N. M. Greene, *XSDRN: A Discrete Ordinates Spectral Averaging Code*, ORNL-TM-2500 (July 1969).

Computations were performed for both the initial and final 6-year fuel cycles for a converter reactor concept that had been analyzed with ROD. The first cycle was of particular interest, because, for the postulated fueling strategy, major changes occurred in the mix of fissile nuclides which affected both the fission-product yields and the reactor neutron energy spectrum. The last cycle was used as a representative example for fission-product buildup in a reactor fueled primarily with ^{233}U . For both cases, the more detailed CITATION calculations essentially duplicated, in all important respects, the fission-product poisoning predicted from the ROD calculations. The results added considerably to the confidence in ROD predictions of fuel-conversion performance. In a similar way, the results add confidence to design studies made with ROD for breeding systems.

1.1.3 Xenon Behavior

J. R. Engel

The program MSRXP developed by H. A. McLain and L. W. Gilley was used by McLain to estimate the xenon-poison fraction in the reference design MSBR for one set of reactor parameters, but with revised nuclear data for the mass-135 nuclides. For coated graphite with a xenon diffusion coefficient of 2.6×10^{-9} cm²/sec in the coating and 2.6×10^{-6} cm²/sec in the bulk material, the estimated poison fraction was revised from 0.0040 to 0.0047. Some uncertainty still remains for this value; the uncertainty is due partly to uncertainties in nuclear data and partly to lack of detailed experimental verification of the model itself. However, the target xenon-poison fraction of <0.005 apparently can be achieved with xenon stripping in a ^{233}U -fueled system if low-permeability graphite is available.

The mathematical model and the computer program for this work have been described.¹⁰

1.2 TRITIUM BEHAVIOR IN MOLTEN SALT SYSTEMS

G. T. Mays

1.2.1 Reactor Calculations

Prior to the interruption of the MSR Program, calculations were made and internally documented

10. H. A. McLain and L. W. Gilley, *The MSRXP (Molten-Salt Reactor Xenon Poisoning) Program*, ORNL-CF-73-2-49 (Feb. 2, 1973).

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1.2.3 Chemical Processing

Continental Oil Company has developed engineered process diagrams and mechanical arrangement drawings for the chemical processing flowsheet supplied by ORNL. This work represents the reduction of the ORNL flowsheet to a practical engineering design in which attention has been given to the details of cooling, heating, shielding, construction, remote maintenance, safety, recovery from accidents and process interruptions, and replacement of major process components. A systematic attempt has been made to include all components and controls that would allow for normal operation. Further, an attempt has been made to utilize the materials that have the lowest possible cost for the service conditions. Specifically, graphite, nickel, and Hastelloy have been utilized in addition to molybdenum.

The process, equipment, and plant layout are described in the Task 1 report⁴ for the flowsheet shown in Fig. 1.13. Continental Oil Company is presently preparing conceptual systems design descriptions for the plant.

1.3 XENON BEHAVIOR IN THE MSBR FUEL SALT SYSTEM

H. A. McLain L. W. Gilley T. C. Tucker

The digital computer program MSRXP (Molten-Salt Reactor XEnon Poisoning) has been written describing in detail the ^{135}Xe behavior throughout the MSBR fuel salt system. This program incorporates the BUBBLE program reported previously⁶ describing in detail the inert purge gas behavior throughout the MSBR fuel salt system. The intent of this effort is to either confirm the results of the simplified calculations of Kedl⁷ describing the noble gas behavior in the MSBR or to make improvements where required.

For the trial case using this program with the following assumptions:

Coated graphite

Number of bubbles: 44 bubbles per cubic centimeter of salt

Total helium dissolved in salt and in gas bubbles: 1.0×10^{-6} moles/cm³

Gas separator efficiency: 90%

the following results were calculated:

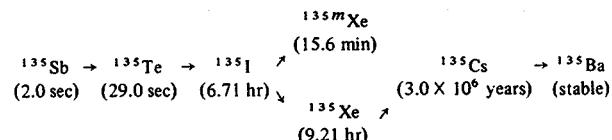
Poison fraction: 0.0038

Average void fraction: 0.0055

Average bubble diameter: 0.0646 cm

Average bubble mass transfer coefficient: 0.0166 cm/sec.

The following mass 135 fission product decay chain along with the indicated half-lives were assumed for the MSRXP program:



We assumed for the purposes of these calculations that the antimony, tellurium, and iodine are "dissolved" in the salt and do not migrate to the bubbles, graphite, or the metal walls. We reasoned that the half-lives of the antimony and tellurium are sufficiently low that the amount of these materials leaving the fuel salt other than by radioactive decay is insignificant. All chemical evidence so far indicates that the iodine remains dissolved in the fuel salt and does not migrate into the graphite. The xenons have a very low solubility in the salt and tend to migrate into the graphite and the gas bubbles in the salt. We assumed that the cesium is stable because of its very long half-life. The yields of ^{135}Sb and ^{135}Xe during the fissioning process have not been established with complete confidence and depend on the fissile material being used. For the initial calculations we assumed that these yields were 5.05 and 1.11% respectively. Also, there is some question about the branching ratio of the ^{135}I decaying to ^{135m}Xe and ^{135}Xe . We assumed that 30% of the ^{135}I decays to ^{135m}Xe in the initial calculations.

Material balances in the form of first-order linear differential equations were written for each of the isotopes present in the fuel salt indicated in the decay chain shown above. Similar material balances were written for the xenons that had migrated into the gas bubbles circulating with the fuel salt. These first-order differential equations were then solved simultaneously throughout the fuel salt loop by an iterative process on the digital computer until a unique solution was obtained.

As was done by Kedl,⁷ we assumed that the rate of migration of the xenons within the graphite and the pyrolytic coating was determined by the Knudsen diffusion of these gases within the graphite. We assumed

6. MSR Program Semiann. Progr. Rep. Feb. 29, 1972, ORNL-4782.

7. MSR Program Semiann. Progr. Rep. Aug. 31, 1968 ORNL-4344, pp. 72-74.

that the migration of ^{135m}Xe and of ^{135}Xe was independent of each other and that the decay of ^{135m}Xe to ^{135}Xe in the graphite did not influence the rate of migration of ^{135}Xe from the fuel salt to the graphite. These isotopes lose their identity in the graphite either by radioactive decay or by reacting with the neutrons in the reactor.

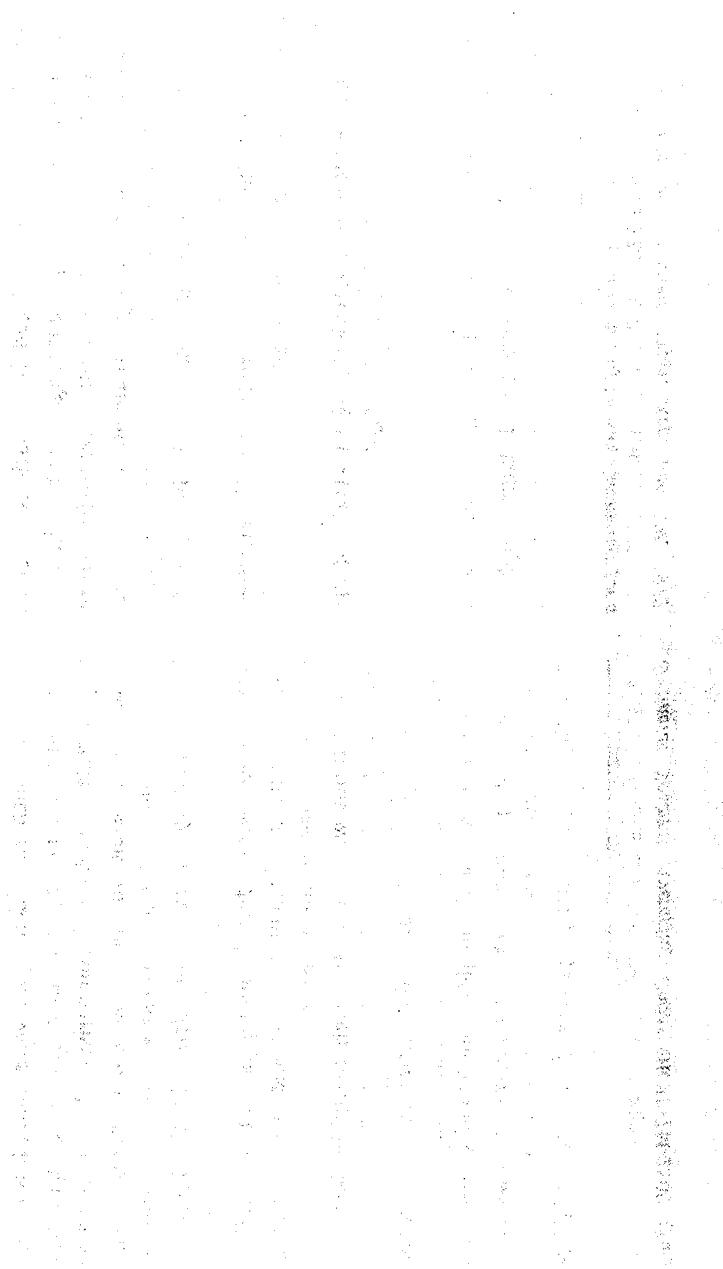
When the unique solution of the material balances for these isotopes is obtained, the distribution of these isotopes throughout the fuel salt loop is then known. From the distribution of ^{135}Xe within the reactor core, the poison fraction due to this isotope, the ratio of the neutrons absorbed in ^{135}Xe to those absorbed in the fissile material, can be determined. We assume that all of the ^{135m}Xe that migrates into the graphite was decayed to ^{135}Xe when we calculate the poison

fraction. This should give a high value for the poison fraction in the reactor.

1.4 HYBRID COMPUTER SIMULATION OF THE MSBR

O. W. Burke

A hybrid computer simulation model of the reference 1000-MW(e) MSBR was developed. The model simulates the plant from the nuclear reactor through the steam throttle at the turbine. The simulation model is being used to determine the dynamic characteristics of the plant as well as to discover the problems associated with the control of the plant. The model and results of some test simulations are described in ORNL-TM-3767, *Hybrid Computer Simulation of the MSBR*.



The diagram illustrates the complex nature of the MSBR's internal components and their interactions. The Nuclear Reactor section shows the reactor core and associated cooling and moderation systems. The Fuel Salt Loop section details the circulation of the sodium-based fuel salt. The Steam Generator section shows the heat transfer between the fuel salt and water to produce steam. The Turbine Generator section shows the conversion of steam energy into electrical power. The Control Room section shows the various sensors and actuators used for monitoring and controlling the plant. The entire system is interconnected by a complex network of pipes, valves, and pumps.

The diagram serves as a key reference for understanding the physical and operational aspects of the MSBR. It provides a clear visual representation of the plant's architecture and the relationships between its various subsystems. The detailed labeling allows for specific components to be identified and analyzed.

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Part 1. MSBR Design and Development

W. R. Grimes P. N. Haubenreich

The design and development program has the purpose of describing the characteristics and estimating the performance of future molten-salt reactors, defining the major problems that must be solved in order to build them, and designing and developing solutions to problems of the reactor plant. To this end we have published a conceptual design for a 1000-MW(e) plant and have contracted with an industrial group organized by Ebasco Services Incorporated to do a conceptual design of a 1000-MW(e) MSBR plant. This design study is to use the ORNL design for background and to incorporate the experience and the viewpoint of industry. One could not, however, propose to build a 1000-MW(e) plant in the near future, so we are doing studies of plants that could be built as the next step in the development of large MSBRs. One such plant is the Molten-Salt Breeder Experiment (MSBE). The MSBE is intended to provide a test of the major features, the most severe operating conditions, and the fuel reprocessing of an advanced MSBR in a small reactor with a power of about 150 MW(t). An alternative is the Molten-Salt Demonstration Reactor (MSDR), which would be a 150- to 300-MW(e) plant based largely on the technology demonstrated in the Molten-Salt Reactor Experiment, would incorporate a minimum of fuel reprocessing, and would have the purpose of demonstrating the practicality of a molten-salt reactor for use by a utility to produce electricity. In addition to these general studies of plant designs, the design activity includes studies of the use of various fuel cycles in

molten-salt reactors and assessment of the safety of molten-salt reactor plants. The fuel cycle studies have indicated that plutonium from light-water reactors has an economic advantage over highly enriched ^{235}U in fueling molten-salt reactors. Some studies related to safety are in progress preliminary to a comprehensive review of safety based on the design of the 1000-MW(e) MSBR.

The design studies serve to define the needs for new or improved equipment, systems, and data for use in the design of future molten-salt reactors. The purpose of the reactor development program is to satisfy some of those needs. Presently the effort is concerned largely with providing solutions to the major problems of the secondary system and of removing xenon and handling the radioactive off-gases from the primary system. Work is well along on the design of the one loop facility for testing the features and models of equipment for the gaseous fission product removal and off-gas systems and for making special studies of the chemistry of the fuel salt. Design is nearing completion and construction is in progress on a second loop facility for studies of equipment and processes and of the chemistry of sodium fluoroborate for the secondary system of a molten-salt reactor. The steam generator is a major item of equipment for which the basic design data are few and the potential problems are many. A program involving industrial participation is being undertaken to provide the technology for designing and building reliable steam generators for molten-salt reactors.

1. Design

E. S. Bettis

1.1 MOLTEN-SALT DEMONSTRATION REACTOR DESIGN STUDY

E. S. Bettis

L. G. Alexander H. A. McLain
C. W. Collins J. R. McWherter
W. K. Furlong R. C. Robertson
H. L. Watts

1.1.1 General

Design and evaluation studies of a 300-MW(e) molten-salt demonstration reactor (MSDR) were continued. This reactor would be a first-of-a-kind prototype to demonstrate the feasibility and to delineate the problems of construction and operation of a large-scale molten-salt reactor power station. Insofar as possible, all aspects of the MSDR design study have been made conservatively and in such a manner as to require development of little new technology. The 26-ft-diam reactor has a sufficiently low power density (~ 10 W/cm³) for the radiation damage to the core moderator to be low enough for the graphite to last the 30-year design life of the plant. The MSDR reactor outlet temperature was taken as 1250°F to increase the margin of safety with regard to radiation damage to Hastelloy N, although it has not been established that this is required. This reduction in temperature, together with use of 900°F steam rather than the 1000°F steam proposed for the MSBR, will allow the nitrate-nitrite salt in the tertiary system to operate at a lower temperature, although, here again, these are not neces-

sarily requirements for an MSBR. It was also decided to operate the steam system at 2400 psia (with reheat to 900°F), again in line with using conservative design conditions for the MSDR. The fuel-salt chemical treatment need not consist of more than fluorination, which is an already well-developed process. Periodic discard of the LiF-BeF₂ carrier salt will be used to limit accumulation of fission products. The built-in safety features of the MSDR give it a "walk-away" capability — that is, the decay heat removal system will operate even with complete loss of station power.

The reactor vessel, the reactor containment and plant building layouts, the fuel-salt storage tanks, and preliminary primary and secondary heat exchanger design concepts have all been described previously.¹⁻³ The design properties of the fuel salt and the secondary and tertiary heat transport salts have been reported.³ During the past report period, however, the configuration of the core graphite was changed, and the method of handling the off-gas and continuously pumping salt back from the drain tank was revised. The drain tank cooling system, along with the drain tank valve and salt catch pan arrangement, was also studied and probably received the major portion of the MSDR design effort during the past period.

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1. *MSR Program Semiannu. Prog. Rep. Feb. 28, 1970, ORNL-4548.*
 2. *MSR Program Semiannu. Prog. Rep. Aug. 31, 1970, ORNL-4622.*
 3. *MSR Program Semiannu. Progr. Rep. Feb. 28, 1971, ORNL-4676.*

1.1.2 Reactor Core

As described previously,² the MSDR reactor vessel is made of Hastelloy N and is about 26 ft in diameter by 35 ft high, including the fuel-salt inlet and outlet plenums at top and bottom. The reactor is graphite moderated and reflected, with the fuel salt circulating upward through the core at a velocity of about 1.5 fps. At this low velocity it will not be necessary to seal the graphite against penetration of xenon in order to maintain the ^{135}Xe poison fraction below 0.003. Use of the unsealed graphite will greatly simplify the production of graphite and will lower the fabrication costs.

The configuration of the core graphite has been changed from that previously reported. Slabs of graphite $1\frac{1}{4} \times 6 \times 9\frac{3}{8}$ in. X about 21 ft long are now spaced

vertically in the core as indicated in the plan view, Fig. 1.1, and in more detail in Fig. 1.2. This configuration is similar to one proposed by the Ebasco Services group in the current industrial study of an MSBR. The methods of assembling the core at bottom and top are shown in elevation in Fig. 1.3. If the slabs are not available in 21-ft lengths, they can be made up of shorter lengths joined together. Because of the relatively low power density in the MSDR, fuel salt trapped in small cracks at the joints would not cause excessive temperatures.

Graphite vertical posts about $2\frac{7}{16} \times 2\frac{7}{16}$ in. in cross section and formed as shown in Fig. 1.2 are used to maintain the configuration of the graphite slabs within the core, while at the same time accommodating thermal and radiation-induced dimensional changes. The lower 6-in. length of each post is turned to $1\frac{1}{2}$ in. in diameter to fit into a hole in the bottom graphite

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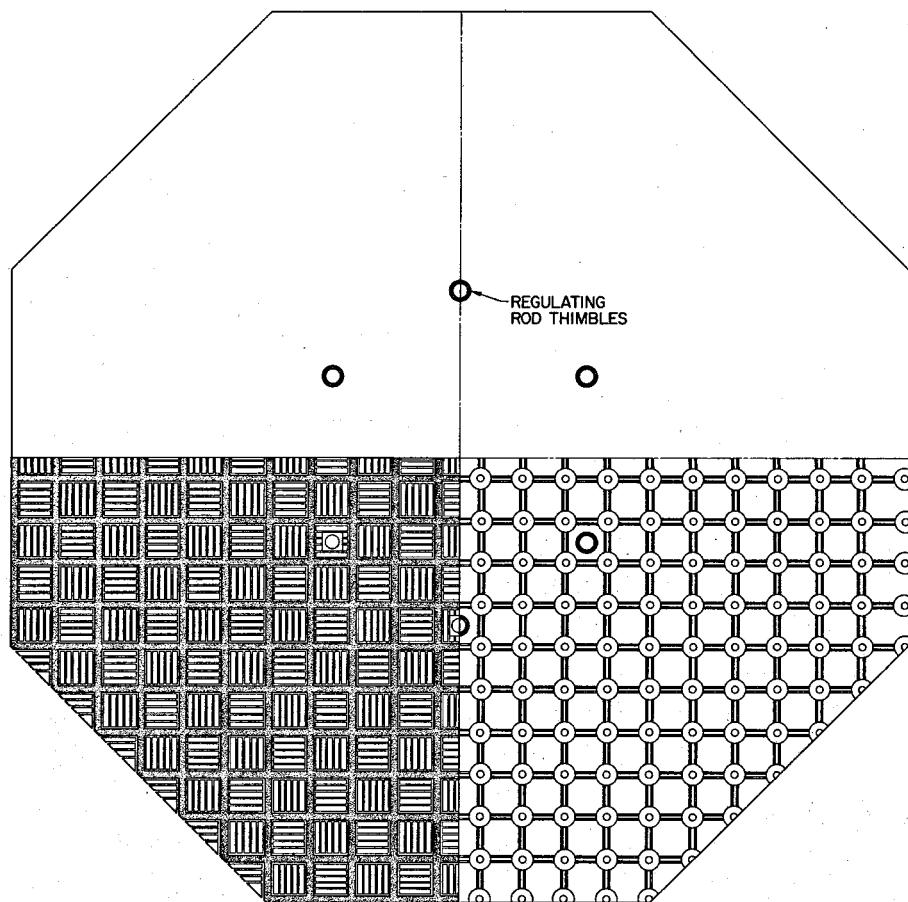


Fig. 1.1. Plan view of 300-MW(e) MSDR reactor core.

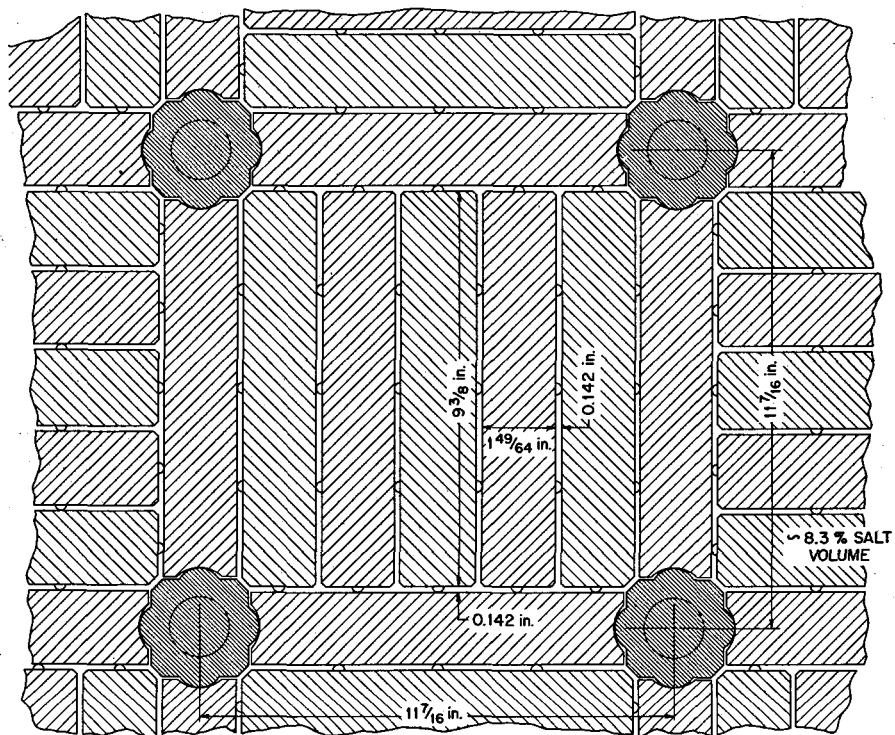


Fig. 1.2. Reactor core cell assembly plan for 300-MW(e) MSDR.

grid plate, and an upper 1-in. length has a diameter of about $\frac{7}{8}$ in. to fit into a corresponding hole in the upper orifice plate, as shown in Fig. 1.3. The posts and slabs thus form a cell about $11\frac{1}{2} \times 11\frac{1}{2}$ in. Dowels provide interstitial spaces for maintaining the upward flow of salt at the design velocity and establish the salt-to-graphite ratio necessary for nuclear performance. Retainer plates at the top of the posts tie the entire core together, as indicated in Fig. 1.3. Individual orifice plates are placed over each cell under the retainer plates to hold down the slabs as well as to provide the necessary orificing of the flow.

The slabs adopted for the MSDR core graphite are easier to fabricate than the prisms used in the MSBR and earlier MSDR designs. The slab shape will also allow the graphite to operate at lower temperatures and at lower stress levels. The graphite temperature at the center of the core has been estimated to be 1190°F.

1.1.3 MSDR Off-Gas and Salt Pump-Back System

The primary function of the fuel-salt drain tank is to store the salt safely when it is drained from the reactor primary system, but the tank serves some secondary purposes as well. The revised MSDR system no longer

depends upon the drain tank as an expansion volume for the salt since the salt-circulation pump bowls are large enough for this purpose. The drain tank continues to serve as a holdup and decay volume for the gases stripped from the circulating fuel salt in normal operation. The drain tank is well suited for this purpose because its heat removal system can handle the approximately 6 MW(t) of heat generated in the gas.

In the MSDR fuel-salt circulation system about 10% of the pump outlet flow is bypassed through a gas separation loop. The salt first passes through a gas separator in this loop, where about 2 cfm (STP) of gas, with 1 to 2 ft³ of salt entrained in it, is removed. Several schemes were studied for separating the entrained salt from the gas, such as spray towers, cyclone separators, etc. The very high heat release from the radioactive gas requires large heat removal systems, however, and the indeterminate manner in which the solid daughters of the gaseous fission products could coat metal surfaces makes it necessary to provide ample surface for accumulation of the particles. After some study it was decided to use the drain tank itself to separate the salt from the gas, the large volume of the tank permitting simple separation by gravity. The large surface of the cooling thimbles in the drain tank would

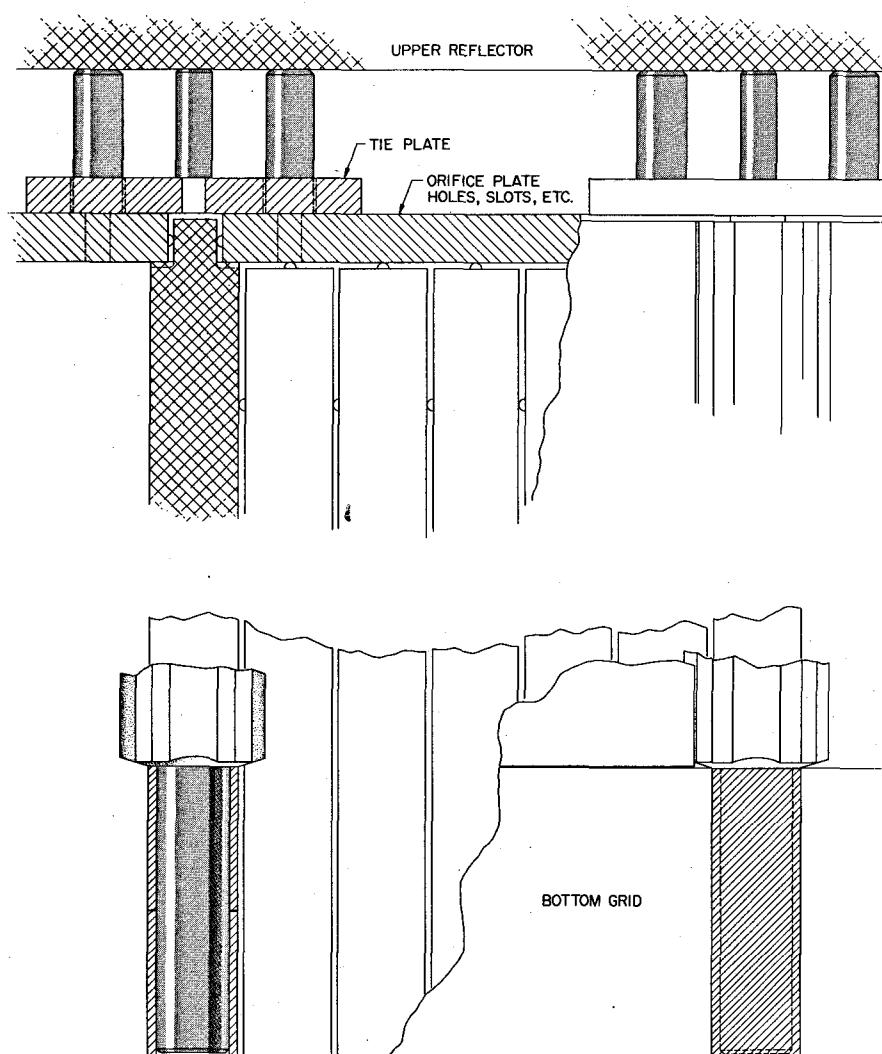


Fig. 1.3. Reactor core cell assembly elevation for 300-MW(e) MSDR.

provide sufficient area for the solid daughters of the noble gases to coalesce.

Each of the gas separators in the three salt-circulation loops in the MSDR discharges the gas-salt mixture into the drain tank through a $1\frac{1}{2}$ -in. line. The temperature of the pipe is limited to about 1325°F by the cooling action of the salt entrained in the gas. The line is provided with a siphon break to prevent drainage of the main salt-circulation loops in the event of a pump failure. The siphon break is in the form of a loop of piping that rises above the level of the liquid in the pump bowl and is equipped with a venturi, the throat of which communicates via a $\frac{1}{2}$ -in. pipe to the gas space above the liquid in the pump bowl. During

normal operation, gas from the bowl is drawn into the venturi and discharged along with the gas-salt mixture from the gas separator into the drain tank. The total flow of salt into the drain tank via the gas separators is estimated to be about 40 gpm. Two small jet pumps in the tank return the salt to the primary loops. The operation of the system is described below, and the design of the jet pumps is described in Sect. 3.7.3.

An elevation of the jet pumps mounted in the center of the drain tank is shown in Fig. 1.4. The assembly is installed so that it can be withdrawn from the tank if repair or replacement becomes necessary. One of the jet pumps takes its suction from the bottom of a vertical 6-in. pipe at the center line of the vessel. The suction

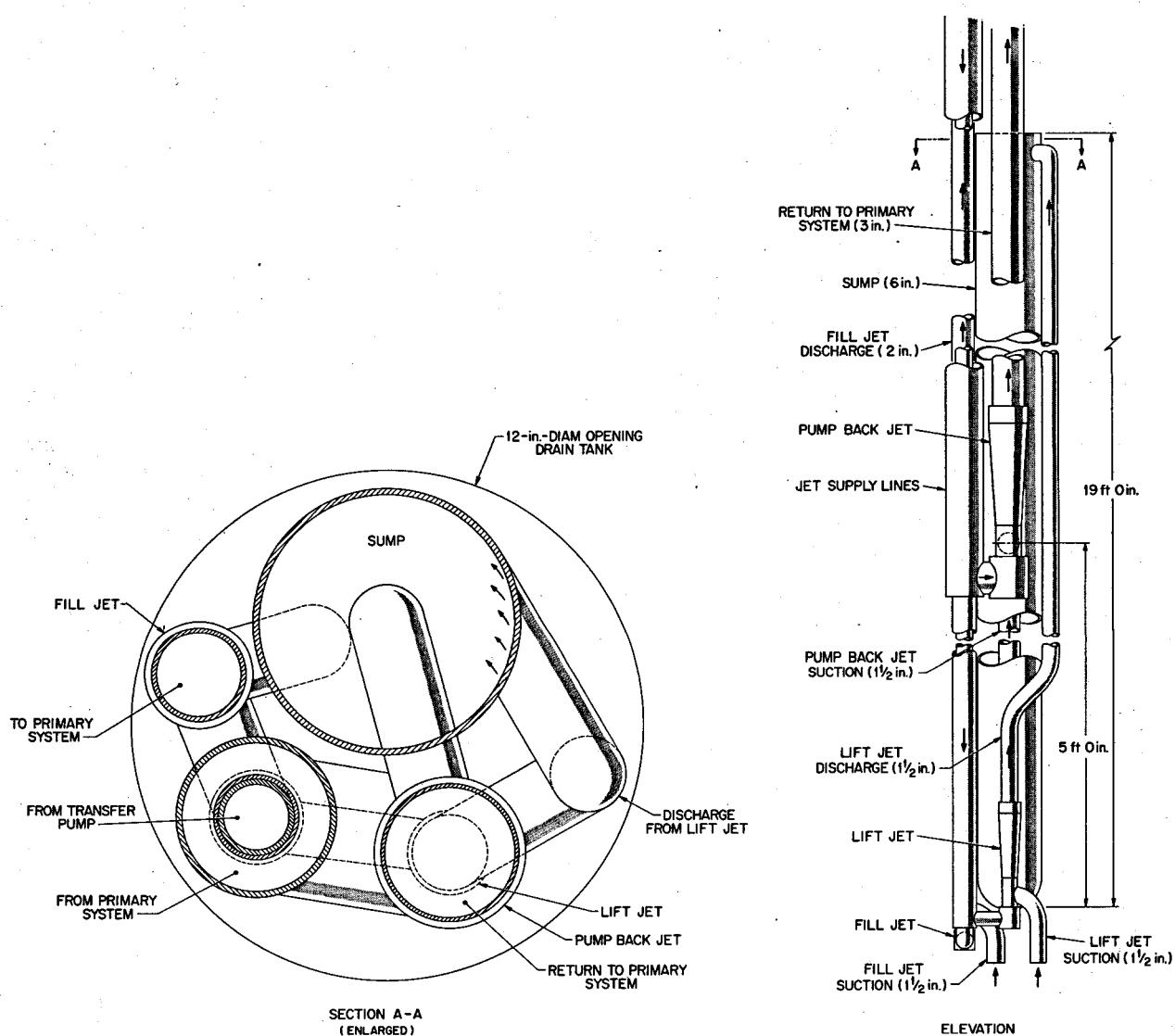


Fig. 1.4. Elevation of jet pump assembly in drain tank.

head on the pump varies from about +15 to -5 ft as the level of salt in the pipe changes from full to empty. Since the discharge from the jet is proportional to the suction head, varying from zero at the lowest level to about 50 ppm when the sump is full, the jet cannot pump gas from the tank when it is empty of salt. The jet pump uses salt as its working fluid, the salt being taken from a manifold connecting the three salt-circulation pumps. Each manifold connection has a ball check valve to prevent backflow in the event of a pump shutdown.

As is also shown in Fig. 1.4, a second jet pump is provided in the assembly to keep the 6-in. central pipe supplied with salt from a small sump in the bottom of the drain tank. This jet has a capacity of about 50 gpm, keeps the liquid pumped out of the drain tank, and, when the tank is empty, may recirculate gas within the tank. As described above, however, the system is designed to prevent pumping of gas from the tank back to the pump bowls. It provides automatic flow control regardless of the number of primary pumps that are running.

A third jet pump is provided in the assembly to fill the primary system with salt from the drain tank and to transfer salt from the drain tank into the fuel-salt storage tank. The working fluid for this third jet pump is provided by a separate small pump located in the chemical processing cell.

1.1.4 Drain Tank Cooling System

The drain tank cooling system described in a previous report² has not been changed in principle, but the thimbles have been rearranged. The change was made to facilitate the manifolding of the pipes which carry the cooling NaK into and out of the thimbles. As shown in the plan view of the top of the drain tank, Fig. 1.5, there are now 60 sets of thimbles, each containing 6 thimbles arranged symmetrically in a circular pattern. Figure 1.6 is an elevation of the drain tank. One of the manifolds is shown in cross section in Fig. 1.6 to indicate that it would be possible to remove and replace manifolds if maintenance were required. By manifolding only six thimbles in a circular pattern, the measures

required to accommodate thermal expansion are made much less stringent. Use of 60 different circuits limits the amount of NaK that could leak from one circuit to about 16 ft³.

In another change in the drain tank system from that described previously,² the one large water tank, to which the NaK circuits reject their heat by radiation, has been replaced by three smaller tanks. The NaK circuits are manifolded in such a manner as to divide the heat removal capability between the three water pools, and with any two water tanks in operation the temperature in the drain tank could be limited to about 1400°F. This arrangement is considered to be more reliable than the previous one in which a single leak could result in total loss of water.

Studies have been initiated on the effects of seismic disturbances on the drain tank and its cooling system. Preliminary calculations indicate that the individual thimbles in the drain tank would have a natural frequency of about 1.5 cps when each thimble is considered to be a 20-ft-long cantilevered member. Since earthquakes produce excitation in the range of 3

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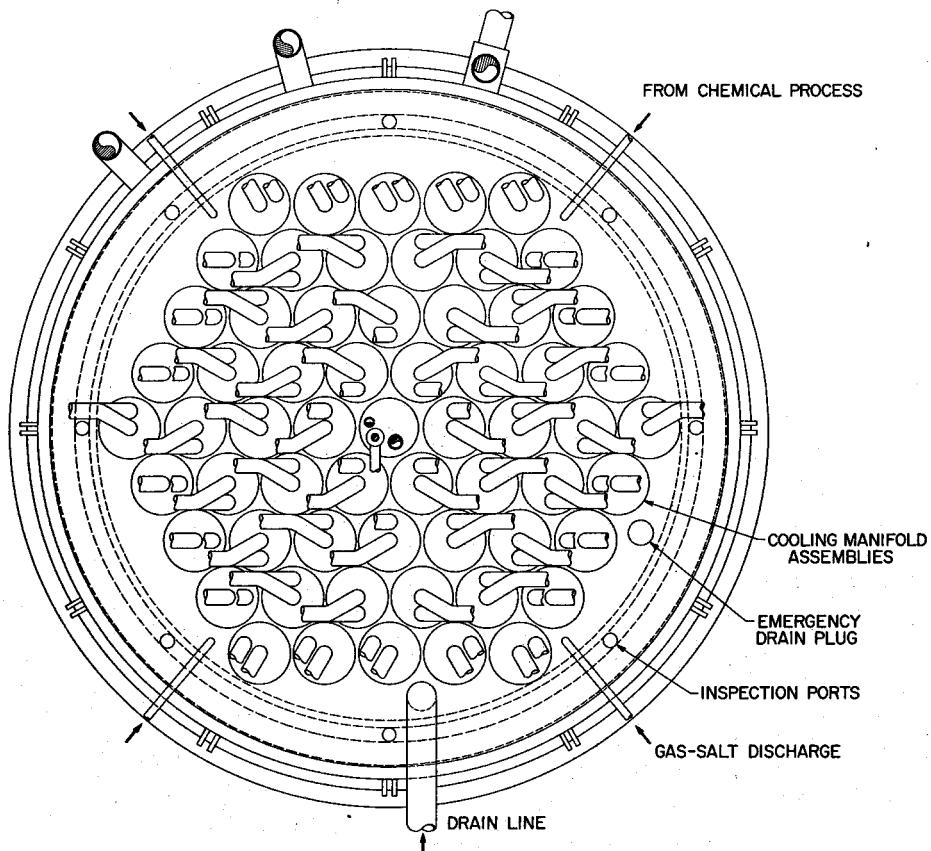


Fig. 1.5. Plan view of top of drain tank showing new thimble arrangement.

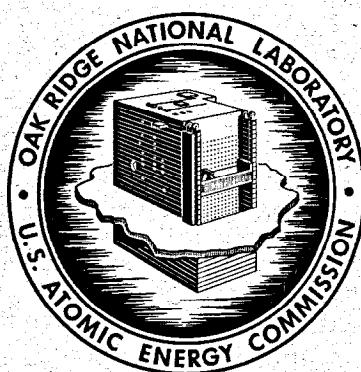
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ORNL-4658
UC-80 – Reactor Technology

CHEMICAL ASPECTS OF MSRE OPERATIONS

Roy E. Thoma



OAK RIDGE NATIONAL LABORATORY
operated by
UNION CARBIDE CORPORATION
for the
U.S. ATOMIC ENERGY COMMISSION

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with the MSRE in power operations showed that oil vapors transported to restricted areas could certainly polymerize (see Sect. 11.1). We had no foreknowledge as to whether polymerization of the recirculated oil might occur, although we faced this prospect with trepidation because of the formation of varnishes from oil vapors in the off-gas valves. This fear seems to have been unwarranted, for throughout the operation of the MSRE the properties of the oil apparently remained unchanged, both as judged on the basis of performance of the components in which it was used as well as from the results of the analyses listed in Table 10.7.

On one occasion, at the time when hydrocarbon varnishes were plugging the off-gas stream valves, the lubricating oil was fractionally distilled and the refractive index of each 10% volume cut was measured. The range was found to include refractive indices from 1.4726 to 1.4817, as compared with the refractive index of the undistilled oil of 1.4738.

Surveillance analysis included infrared absorption tests to determine whether oxidation was occurring. Occasionally these tests indicated that C-O bands were present, but these bands were only slightly more intense than in the reference sample and seemed not to become more so with continued use of the oil.

Among the least well-understood chemical phenomena governing behavior in the MSRE and, as well, among the least tractable for experimental investigation with the MSRE were the chemical relationships involving oil from the MSRE fuel pump and the transport and distribution of fission products and tritium within the reactor system. The properties of the oil, as determined through the chemical monitoring program, disclosed little that was significant concerning these matters. Unquestionably, the thermal degradation products formed in the pump bowl were of some significance in the transport behavior of fission products and tritium. Resolution of their exact effects on reactor operations, however, will be derived from future investigations rather than from further scrutiny of MSRE behavior.

References

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2. R. C. Robertson, *MSRE Design and Operations Report, Part I*, ORNL-TM-728, p. 444.
3. R. C. Robertson, *ibid.*, p. 144.
4. A. G. Grindell, *Specification of a Lubricant for MSRE Pumps*, ORNL-MSR-60-37.

S. A. G. Grindell and P. G. Smith, *MSR Program Semiannual Progr. Rep. July 31, 1964*, ORNL-3708, p. 155.

11. TRANSPORT OF MATERIALS FROM SALT TO COVER GAS SYSTEMS

11.1 Fission Products

In solid fuel reactors, knowledge of the modes of fission product distribution is of utmost importance to the assessment of reactor safety and control and for the development of reprocessing methods. Changes in state and distribution of fission product species in solid fuel matrices have been examined so intensely and for so long a period of time that only problems related to accident conditions remain as challenging to the understanding. For molten-salt reactors it is necessary that an even more comprehensive understanding of fission product behavior be achieved, for with fission products intentionally circulated with fuel salt, their disposition, as controlled by mechanical and chemical influences, is of considerably greater importance than in static fuel systems. In MSRs, some fission products are removed from the fuel salt continuously and become distributed to the off-gas system, to the fuel storage tanks, and to the processing plant. They must be contained and their decay heat removed under all conceivable circumstances.

The probable fates of fission products produced in molten-salt reactors were deduced from thermodynamic considerations, from laboratory studies, and from in-pile static capsule tests. From the results of these studies, one may group fission product elements in categories according to general chemical properties: noble gases: Kr, Xe; noble metals: Pd, Rh, Ru, Mo, Ag, Te, Tc, Nb, Se, As; transition metals: Sn, Sb, In, Cd, Ga, Zr, Zn; lanthanides: Y, La, Ce, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho, Er; active metals: Cs, Rb, Ba, Sr; halides: Br, I.

Of the elements produced in high yields, only the lanthanides developed concentrations in the MSRE that were in the range of chemical detection. This group accounts for 53.8% of the fission products. Operation of the MSRE resulted in the fissioning of 2.88 kg of ^{235}U fuel and 1.32 kg of ^{233}U , a total of 17.92 g-atoms of uranium. From fission, therefore, 9.64 g-atoms of lanthanides, about 1.5 kg, were produced ultimately to produce a final concentration of 300 ppm of these elements in the fuel salt. At the very low concentrations which prevailed in the fuel salt, the

lanthanides were of little consequence with respect to general chemical surveillance, and no special effort was made to monitor their concentrations or distributions directly. One method for their removal from MSR carrier salts relies on high-temperature distillation. Tests of this separation method with MSRE fuel carrier salt disclosed that the measured concentrations of the lanthanides in the salt agreed well with computed values.¹

The MSRE provided the first opportunity to begin studies of fission product transport in circulating salt systems. Although some preliminary information was derived from the results of experiments with the Aircraft Reactor Experiment, its scheduled period of operation was so brief as to obviate any real opportunity for such studies. Recognition that the knowledge of fission product behavior was in a rudimentary condition required that programs for their investigation proceed independently from that for general chemical surveillance. Detailed results of the studies of fission product behavior are therefore excluded from the present report and will be described separately.²

11.2 Restrictions in the Off-Gas System

In both the fuel and coolant systems, control of the pressure in the cover-gas systems over extended periods proved to be difficult and unpredictable. The cause was the accumulation of solids and tarry deposits, mostly originating from lubricating oil, in the off-gas throttling valve used for fuel-system pressure control and in the filter just upstream of the coolant-system pressure-control valve.

During the initial period of circulation of coolant salt in the reactor, a period of 1200 hr, the off-gas filter plugged and was replaced twice. Examination of the filter medium showed that it was covered with amorphous carbon containing traces of the constituents of the coolant salt and Hastelloy N. Although the filter was replaced by another one with some 35 times the surface area, pressure control again became erratic. It was realized that oil seepage took place down the shaft of both the fuel pump and the coolant pump. Rates of seepage were determined and found to be variable, ranging from indicated rates of zero to 4 cm³ of oil per day seeping into the pump bowl.

The rates did not seem to increase appreciably from time to time, nor was their variation clearly related to any operational practice. It became routine to replace the filter in the coolant salt off-gas system after several months' use. Each time the filter was replaced, it was found to be covered with oil residues.

Pressure control of the fuel system became erratic toward the end of the first extended operation with salt in the circuit. During this period, ⁸⁵Kr was used for tests of the stripping efficiency of the noble gas from the helium stream. After saturation of the salt to near equilibrium concentrations, the inflow of ⁸⁵Kr was stopped and the gas purged from the system. As one part of the tests, the effect of salt level in the pump bowl was examined with the LiF-BeF₂ (66-34 mole %) salt at three levels. At this time the void fraction of the salt in the fuel loop was measured for the first time and found to be 0.6%. At termination of these tests the off-gas filter was removed and examined. It was found to be free of deposits. Examination of the pressure control valve showed that it was partially plugged. The residue was washed out with acetone. The liquid suspension was dark in color, and the residue left after evaporation of the acetone consisted of spheres of a glassy material, 1 to 5 μ in diameter. Shortly after the cleaned control valve was reinstalled, it became partially restricted again. It was then replaced with another one of greater flow-through capacity. Inspection of the original valve revealed that a black deposit partially covered the tapered stem. The deposit was about 20% amorphous carbon and the remainder was again 1- to 5- μ spheres that were found to have the composition of the flush salt.

The program for bringing the MSRE to full-power operation was interrupted on two occasions by flow restrictions in the off-gas system. Symptoms of plugging became pronounced in each case after about 12 hr of operation at 1 MW. Restrictions were first noted in the capillary restrictor in line 521; at the same time, check valve HCV-533 became inoperative. Restriction was also noted in the stainless steel line filter and 522 valve assembly. In order to resume operation of the reactor, the capillary restrictor and HCV-533 check valve were replaced by nonrestrictive sections, and the filter-valve assembly was replaced. The filter element which was removed was capable of blocking passage of more than 90% of the particulate matter of $\geq 0.7 \mu$ in diameter. It was replaced by a filter designed to block particulates of $< 50 \mu$ in diameter.

Continued difficulty was experienced with restrictions in the off-gas system of the fuel circuit for some time during the early stages of power operation with the MSRE. These difficulties led to a practice which relieved excess pressure in the fuel system by venting gas through HCV-533 to the auxiliary charcoal bed. During this period, evidence of restrictions in the charcoal beds began to appear. On resuming operation at 1 MW, restrictions appeared to develop in three

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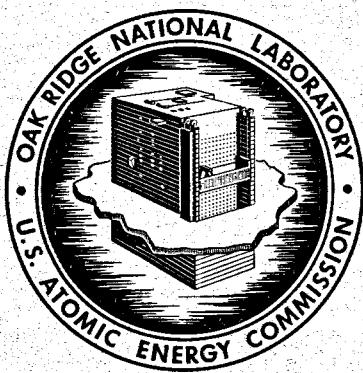
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ORNL-4622
UC-80 — Reactor Technology

MOLTEN-SALT REACTOR PROGRAM

SEMIANNUAL PROGRESS REPORT

For Period Ending August 31, 1970



OAK RIDGE NATIONAL LABORATORY
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5. Systems and Components Development

Dunlap Scott

The reorientation of the MSBR program into a technology program has resulted in curtailment of efforts to produce components for the MSBE and added emphasis on a program of experiments and studies that will explore more deeply some of the technical problems and their significance to the MSBR design. The program for the participation of industry in the design and fabrication of steam generators has been reduced to provide for design studies only. The programs concerned with further study of the technological problems will center around the operation of a gas system test loop, a secondary system technology loop, and later a steam generator technology loop.

The gas system test loop will be constructed of Hastelloy N and will be operated with a salt containing lithium, beryllium, thorium, and uranium in proportions representative of the fuel salt for a single-fluid MSBR. The loop will be used to evaluate the operation of the gas-removal systems under different chemical states of the salt as well as to investigate other technical areas of interest to the fuel system. We expect to improve our confidence in the use of water as an easy fluid for developing components which are actually to be operated in salt.

The secondary system technology loop, also constructed of Hastelloy N, will be operated with a sodium fluoroborate eutectic to study the mechanical and chemical features of the secondary salt system. Other coolants may be tested in the loop in search of alternatives to sodium fluoroborate. This loop will also provide salt for circulation through the steam generator technology loop, which will be operated to study basic problems of molten-salt-heated steam generators.

The program for the procurement and testing of pumps for an MSBE has been curtailed until the program for the construction of the MSBE is reactivated.

The program for the development of equipment and techniques for remote welding of pipe and vessel

closures has been centered around establishing the feasibility of an automatic welding system which, by virtue of its reliability and versatility, would be adaptable to remote operation. The interest in this equipment has grown to such an extent that the results of this feasibility study probably will be perfected by others for use in making field welds during construction. Our program will continue with a study of such associated considerations as the remote inspection of weld passes and remote preparation, cleanup, and alignment for welding. The status of the programs which have been started is given below.

5.1 GASEOUS FISSION PRODUCT REMOVAL

C. H. Gabbard

5.1.1 Gas Separator

Water testing of the latest gas separator design¹ has continued. Although the attachment of the gas core to the recovery vane hub is much improved over the previous designs, the main vortex still tends to be somewhat unstable in front of the recovery hub at low gas flow rates, as shown in Fig. 5.1. At higher gas flow rates the vortex is more stable and becomes fully attached to the recovery hub, as shown in Fig. 5.2. Under both of these conditions, some of the gas escapes the vortex and bypasses the separator. Although all the gas bubbles appear to move to the central vortex, losses around the recovery hub reduce the actual gas removal efficiency to less than 100%.

Performance data for the gas separator have been taken at flow rates of 440, 570, and 660 gpm of water with gas input rates ranging up to 11.0 scfm. The maximum efficiency in each test occurred at the highest

¹MSR Program Semiann. Progr. Rept. Feb. 28, 1969, ORNL-4396, p. 96.

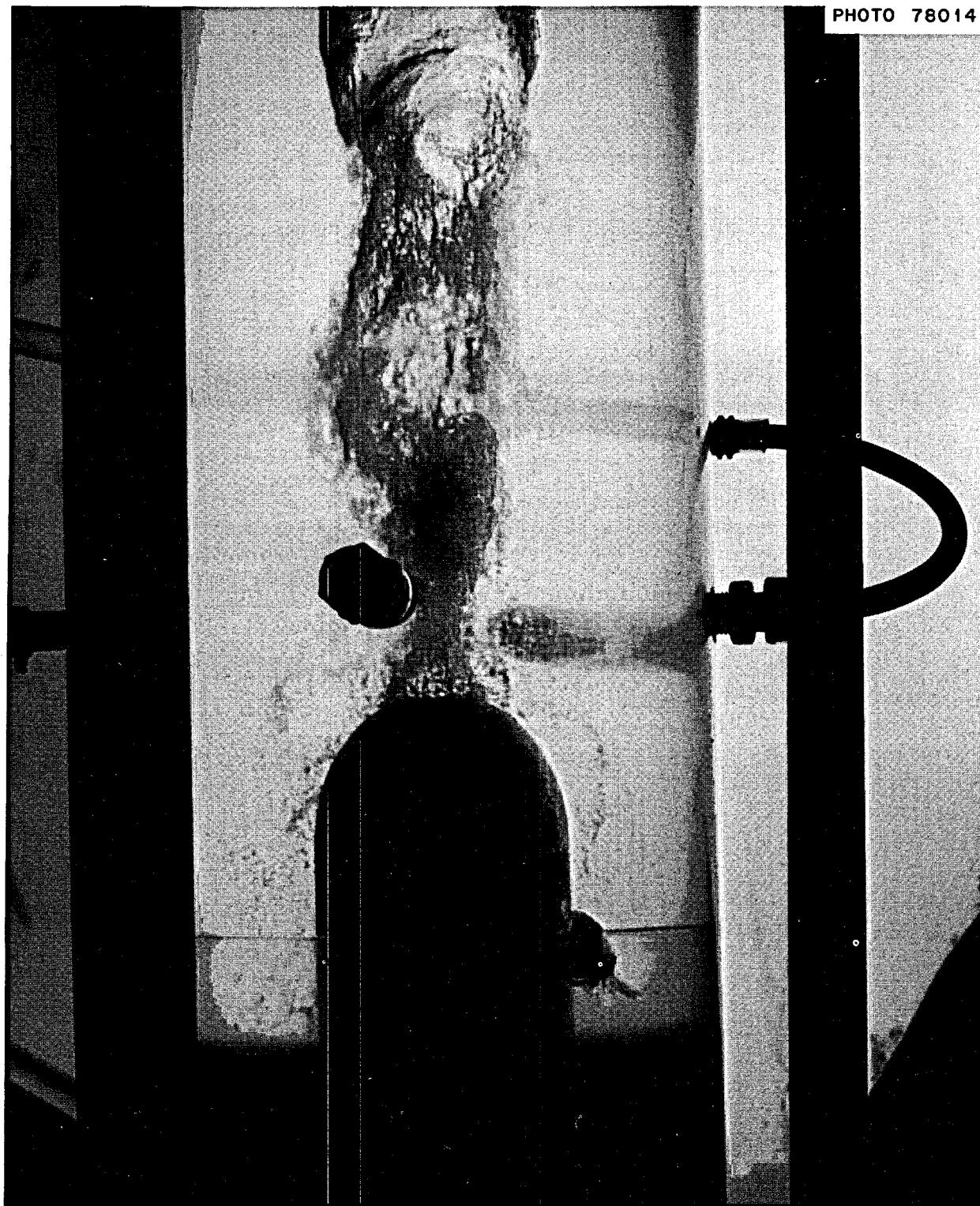


Fig. 5.1. Gas Separator Recovery Hub Showing Unstable Vortex.

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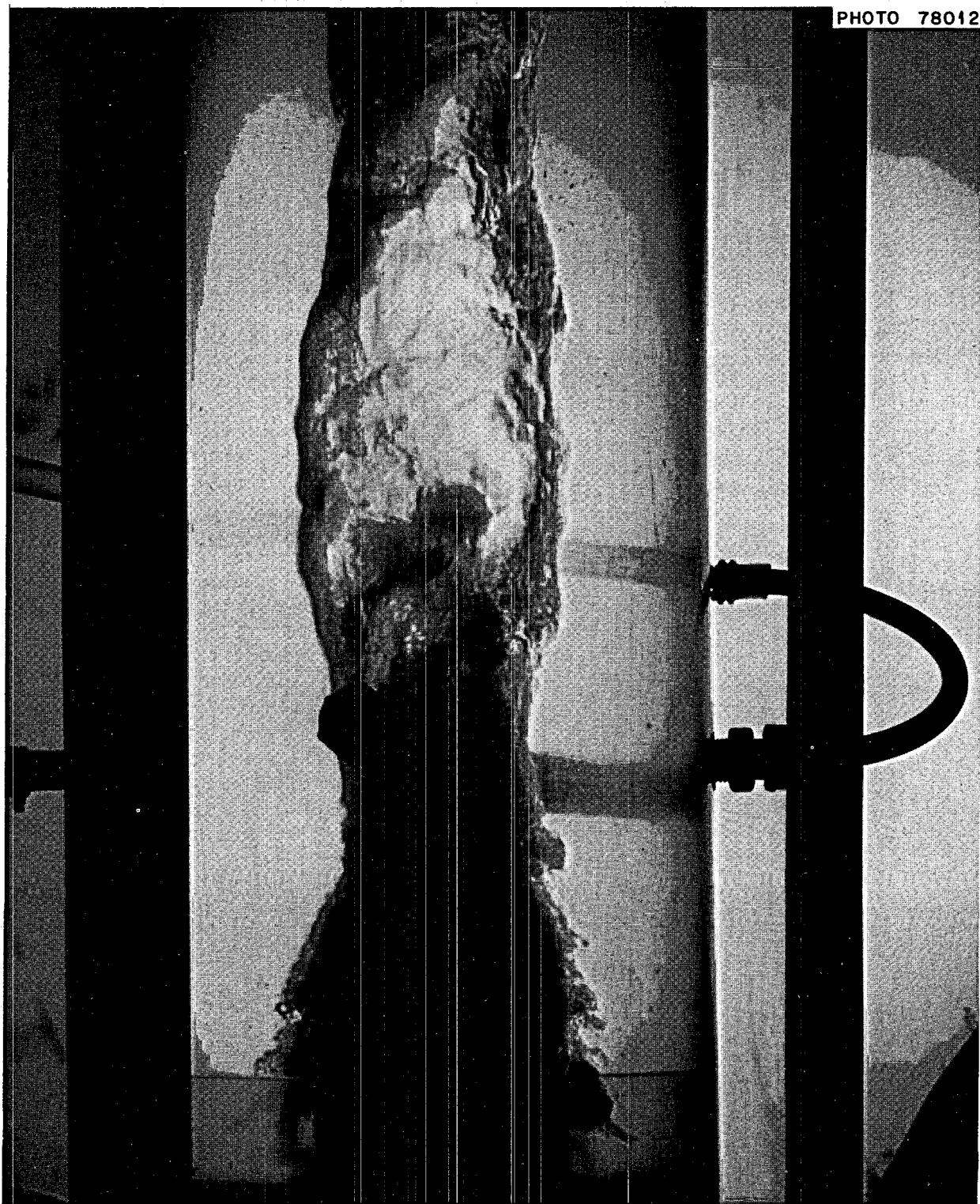


Fig. 5.2. Gas Separator Recovery Hub Showing Fully Attached Vortex.

gas input rate and was nominally 94, 88, and 78% for the three water flow rates respectively. The higher efficiency at the lower water flow rate is probably the result of reduced turbulence at the recovery hub. The results of the performance tests at 570 gpm are shown in Fig. 5.3. Each of the three curves was taken with a constant throttle setting on the gas-water takeoff line. The inlet void percentage for the MSBE separator, based on an average of 0.2% voids in the loop, would be about 0.1% but would vary depending on the operating pressure of the separator. Although the removal efficiency increased with increasing gas flow rate up to 1.0 scfm, the gas removal capacity of the separator is limited, and the removal efficiency would fall off rapidly at somewhat greater gas flow rates.

The recovery hub of the separator is being redesigned with an annular gas takeoff port in an attempt to improve the vortex stability and to increase the removal efficiency at the lower void fractions.

5.1.2 Bubble Generator

The design and testing of a reduced-scale "teardrop" bubble generator was presented previously.² The tests reported at that time indicated that the bubble diam-

eter was about one-fourth of the flow annulus thickness and that an annulus of about 0.080 in. would be required to produce the desired 0.020-in.-diam bubbles. A multiple vane design was envisioned for the full-sized MSBE or MSBR bubble generator to avoid the large diameter that would be required for a "teardrop" design with the 0.080-in. flow annulus. A reduced-scale prototype model with a single vane is shown in Fig. 5.4. A satisfactory bubble size was generated by this design, but there were flow oscillations around the trailing edge of the vane in the vertical plane and also in a horizontal plane within each of the diffuser sections. There was also difficulty in distributing the gas flow uniformly across the width of the flow channels and between the top and bottom gas feed holes.

The development tests of both the "teardrop" and the "multivane" bubble generators have been discontinued for the present time to evaluate a simpler "venturi" design. Two commercial jet pumps with throat diameters of 0.316 and 0.656 in. were used for the preliminary testing. The tests indicated that a satisfactory bubble size could be obtained in water at throat velocities greater than about 30 fps. The bubble size did not appear to be pressure dependent between throat pressures of 1.47 and 2.9 psia. The only difficulty encountered with the jet pumps was pulsations in the bubble output. The pulsations occurred when the jet pumps were run at throat pressures greater than about 1 to 3 psia. The suction cavity of the larger jet pump was filled with epoxy and remachined with a minimum gas volume in an attempt to eliminate the pulsations. The pulsations were reduced but were not eliminated by this change.

A larger pressure drop across the gas feed ports will probably be required to obtain a constant bubble output under all operating conditions.

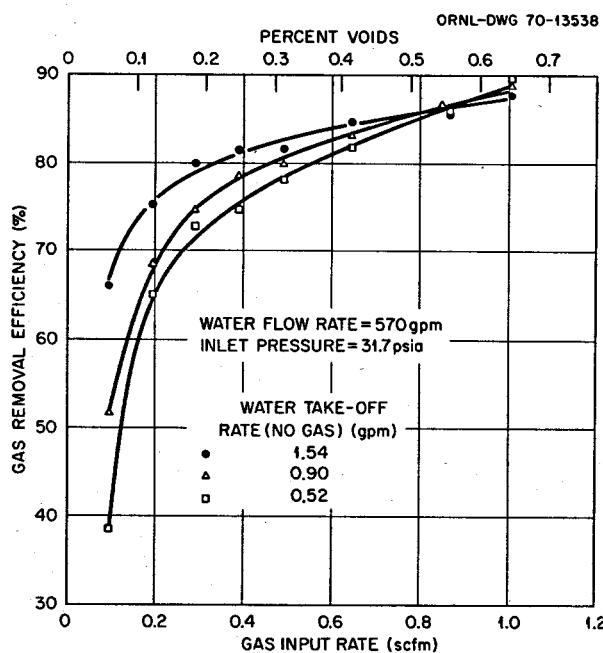


Fig. 5.3. Performance of 4-in-ID Gas Separator with 0.302-in.-diam Gas Removal Port.

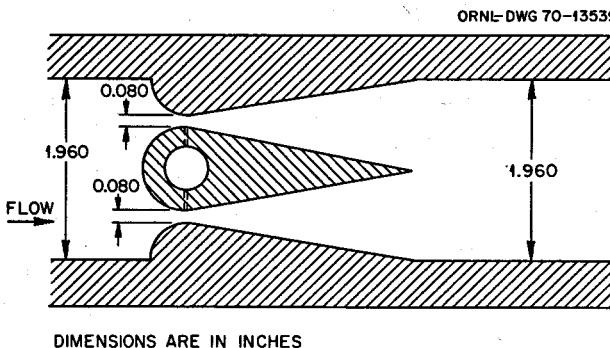


Fig. 5.4. Section Through Prototype Multivane Bubble Generator.

Part 3. Chemistry

W. R. Grimes

The chemical research and development activities described below are conducted to establish the basic chemical information required for the development of advanced molten-salt reactor systems.

A substantial fraction of these efforts continued to be devoted to the transport, distribution, and chemistry of fission products in the MSRE. Investigations of fission product behavior have been continued with specimens removed from the MSRE fuel circuit, with "synthetic" fuel mixtures, and by investigation of the chemistry of molybdenum, niobium, and ruthenium in molten fluoride mixtures.

A broad program of fundamental investigations into the physical chemistry of molten-salt systems was maintained; from it are derived the basic data for reactor and chemical reprocessing design. Within the scope of these efforts are included research in solution thermodynamics and phase equilibria, crystal chemistry, electrochemistry, spectroscopy (both Raman and electronic absorption), transport processes, and theoretical aspects of molten-salt chemistry.

Studies of the chemical aspects of separations methods were continued. The results of these studies

form the basis for evolving modifications of methods for reprocessing MSBR fuel salts. With recent adoption of a fundamental change in the reference design for MSBR fuel reprocessing, one which effects transfer of rare earths from liquid bismuth to lithium chloride as an acceptor solvent, emphasis in separation studies has expanded to include the chemistry of molten chloride systems.

The principal emphasis of analytical chemical development programs has been placed on methods for use in semiautomated operational control of molten-salt breeder reactors, for example, the development of in-line analytical methods for the analysis of MSR fuels, for reprocessing streams, and for gas streams. These methods include electrochemical and spectrophotometric means for determination of the concentration of U^{3+} and other ionic species in fuels and coolants and adaptation of small on-line computers to electroanalytical methods. Parallel efforts have been devoted to the development of analytical methods related to assay and control of the concentration of water, oxides, and tritium in fluoroborate coolants.

8. Fission Product Behavior

8.1 NOBLE METAL FISSION PRODUCT BEHAVIOR

E. L. Compere E. G. Bohlmann

It was noted in the previous report¹ that noble metals in MSRE salt samples acted as if they were particulate

constituents of a mobile "pool" of such substances held up in the system for a substantial period and that evidence regarding this might be obtained from the activity ratio of pairs of isotopes.

Pairs of the same element, thereby having the same chemical behavior (e.g., $^{103}\text{Ru}/^{106}\text{Ru}$), should be particularly effective. As produced, the activity ratio of such a pair is proportional to ratios of fission yields and decay constants. Accumulation over the operating history yields the inventory ratio, ultimately propor-

¹E. L. Compere and E. G. Bohlmann, *MSR Program Semiann. Progr. Rept. Feb. 28, 1970*, ORNL-4548, pp. 111-18.

tional (at constant fission rate) only to the ratio of fission yields. If however there is an intermediate holdup and release before final deposition, the activity ratio of the retained material will depend on holdup time and will fall between production and inventory values. Furthermore, the material deposited after such a holdup will, as a result, have ratio values lower than inventory. (Values for the isotope of shorter half-life, here ^{103}Ru , will be used in the numerator of the ratio throughout our discussion.) Consequently, comparison of an observed ratio of activities (in the same sample) with associated production and inventory ratios should provide an indication of the "accumulation history" of the region represented by the sample. Since both determinations are for isotopes of the same element in the same sample (consequently subjected to identical treatment), many sampling and handling errors cancel and do not affect the ratio. Ratio values are thereby subject to less variation.

We are in process of using the $^{103}\text{Ru}/^{106}\text{Ru}$ activity ratio among others to examine samples of various kinds taken at various times in the MSRE operation. These include salt and gas samples from the pump bowl and other materials briefly exposed there at various times.

Data are also available from the sets of surveillance specimens removed after runs 11,² 14,³ 18,⁴ and 20.⁵ Materials removed from the off-gas line after runs 14⁶ and 18⁷ offer useful data. Some information is available⁸ from the on-site gamma spectrometer surveys of the MSRE following runs 18 and 19, particularly with regard to the heat exchanger and off-gas line.

Inventory and Model. — The data will be discussed in terms of a "compartment" model, which will assign first-order transfer rates common for both isotopes between given regions and assume this behavior was consistent throughout MSRE history. Because the

half-lives of ^{103}Ru and ^{106}Ru are quite different, 39.6 days and 367 days respectively, appreciably different isotope activity ratios are indicated for different compartments and times as simulated operation proceeds. A sketch of a useful scheme of compartments is shown in Fig. 8.1.

We assume direct production of ^{103}Ru and ^{106}Ru in the fuel salt in proportion to fission rate and fuel composition as determined by MSRE history. The material is fairly rapidly lost from salt either to "surfaces" or to a mobile "particulate pool" of agglomerated material. The pool loses material to one or more final repositories, nominally "off-gas," and also may deposit material on the "surfaces." Rates are such as to result in an appreciable holdup period of the order of 50 to 100 days in the "particulate pool." Decay, of course, occurs in all compartments.

Material is also transferred to the "drain tank" as required by the history, and transport between compartments ceases in the interval.

From the atoms of each type at a given time in a given compartment, the activity ratio can be calculated, as well as an overall inventory ratio.

We shall identify samples taken from different regions of the MSRE with the various compartments and thus obtain insight into the transport paths and lags leading to the sampled region. It should be noted that a compartment can involve more than one region or kind of sample. The additional information required to establish the *amounts* of material to be assigned to a given region, and thereby to produce a material balance, is not available.

In comparison with the overall inventory value of $^{103}\text{Ru}/^{106}\text{Ru}$, we should expect "surface" values to equal it if the deposited material comes rapidly and only from "salt," and to be somewhat below it if in addition "particulate" is deposited. If there is no direct deposition from "salt" to "surface," but only "particulate," then deposited material should approach "off-gas" compartment ratio.

The "off-gas" compartment ratio should be below inventory, since it is assumed to be steadily deposited from the "particulate pool," which is richer in the long-lived ^{106}Ru component than production, and inventory is the accumulation of production minus decay.

The particulate pool will be above inventory if material is transferred to it rapidly and lost from it at a significant rate. Slow loss rates correspond to long holdup periods, and ratio values tend toward inventory.

Differential equations involving proposed transport, accumulation, and decay of ^{103}Ru and ^{106}Ru atoms

²S. S. Kirslis and F. F. Blankenship, *MSR Program Semiann. Progr. Rept. Aug. 31, 1967*, ORNL-4191, pp. 121-28.

³S. S. Kirslis and F. F. Blankenship, *MSR Program Semiann. Progr. Rept. Aug. 31, 1968*, ORNL-4344, pp. 115-41.

⁴F. F. Blankenship, S. S. Kirslis, and E. L. Compere, *MSR Program Semiann. Progr. Rept. Aug. 31, 1969*, ORNL-4449, pp. 104-7; F. F. Blankenship, S. S. Kirslis, and E. L. Compere, *MSR Program Semiann. Progr. Rept. Feb. 28, 1970*, ORNL-4548, pp. 104-10.

⁵F. F. Blankenship, personal communication.

⁶E. L. Compere, *MSR Program Semiann. Progr. Rept. Aug. 31, 1968*, ORNL-4344, pp. 206-10.

⁷E. L. Compere and E. G. Bohlmann, *Fission Product Behavior in MSRE During ^{233}U Operation*, ORNL-TM-2753 (in preparation).

⁸A. Houtzeel, personal communication.

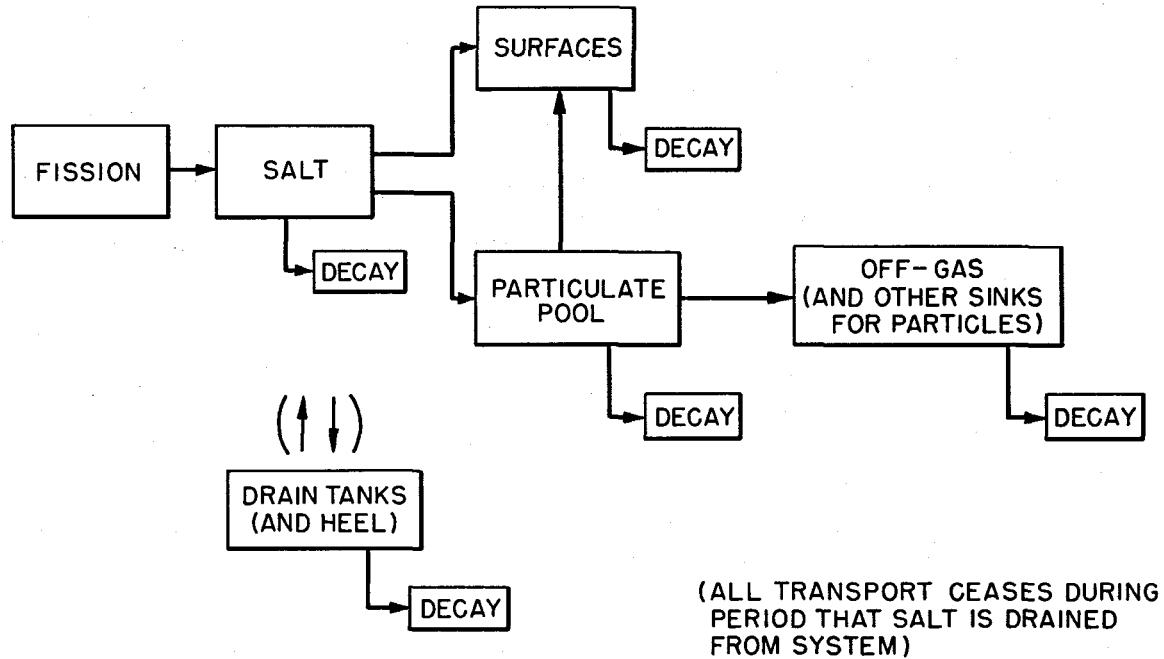


Fig. 8.1. Compartment Model for Noble-Metal Fission Transport in MSRE.

with respect to these compartments were incorporated into a fourth-order Runge-Kutta numerical integration scheme which was operated over the full MSRE power history.

The rates used in one calculation referred to in the discussion below show rapid loss (<1 day) from salt to particulates and surface, with about 4% going directly to surface. Holdup in the particulate pool results in a daily transport of about 0.2% of the pool to "surface" and about 2% per day of the pool to "off-gas," for an effective average holdup period of about 45 days. All transport processes are assumed irreversible in this scheme.

Off-Gas Line Deposits. — Data have been reported on the examination⁶ after run 14 (March 1968) of the jumper line installed after run 9 (December 1966), of the examination⁷ after run 18 (June 1969) of parts of a specimen holder assembly from the main off-gas line installed after run 14, and of the examination⁷ of parts of line 523, the fuel pump overflow tank purge gas outlet to the main off-gas line, which was installed during original fabrication of MSRE. These data are shown in Table 8.1.

For the jumper line removed after run 14, observed ratios range from 2.4 to 7.3. By comparison the inventory ratio for the net exposure interval was 12.1. If a holdup period of about 45 days prior to deposition in the off-gas line is assumed, we calculate a lower ratio for the compartment of 7.0. It seems indicated that a holdup of ruthenium of 45 days or more is required.

Ratio values from the specimen holder removed from the 522 line after run 18 ranged between 9.7 and 5.0. Net inventory ratio for the period was 19.7, and for material deposited after a 45-day holdup we estimated a ratio of 12.3. A longer holdup would reduce this estimate. However, we recall that gas flow through this line was appreciably diminished during the final month of run 18. This would cause the observed ratio values to be lower by an appreciable factor than would ensue from steady gas flow all the time. A holdup period of something over 45 days still appears indicated.

Flow of off-gas through line 523 was less well known. In addition to bubbler gas to measure salt depth in the overflow tank, part of the main off-gas flow from the pump bowl went through the overflow tank when flow through line 522 was hindered by deposits. The

ORNL-4548
UC-80 - Reactor Technology

Contract No. W-7405-eng-26

**MOLTEN-SALT REACTOR PROGRAM
SEMIANNUAL PROGRESS REPORT**

For Period Ending February 28, 1970

M. W. Rosenthal, Program Director

R. B. Briggs, Associate Director

P. R. Kasten, Associate Director

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AUGUST 1970

**OAK RIDGE NATIONAL LABORATORY
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U. S. ATOMIC ENERGY COMMISSION**

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1.2.6 Fission Product Distribution in the Fuel System

A. Houtzeel R. Blumberg F. F. Dyer

Gamma-ray spectrometry studies of fission product distribution in the fuel system were continued during this period with the following progress:

1. Gamma-ray spectra were taken of the main reactor off-gas line (line 522) near the fuel pump with the reactor at different operating conditions.
2. Calibration of the spectrometer equipment with ^{110m}Ag and ^{226}Ra sources was completed.
3. Several components of the MSRE system were scanned during the November shutdown period and after the final reactor shutdown.
4. Work was continued on a computer program to automatically analyze the spectra.

5. Some preliminary analyses were made of the gamma-ray spectra taken during the June-July 1969 reactor shutdown.

The equipment for this work was described previously.¹¹ Figure 1.8 is a photograph of the detector system in operation.

Reactor Off-Gas Line. — The gamma-ray detector, together with the collimator body and insert, was set up over a hole in the reactor shield blocks. Through this hole it was possible to aim the detector directly at the main reactor off-gas line, with only the cell membrane interposed, at a location about 1 ft downstream of the fuel pump. The purpose of this experiment was to study any variation in the fission products contained in

¹¹MSR Program Semiann. Progr. Rept. Aug. 31, 1969, ORNL-4449, pp. 11-12, 31-33.

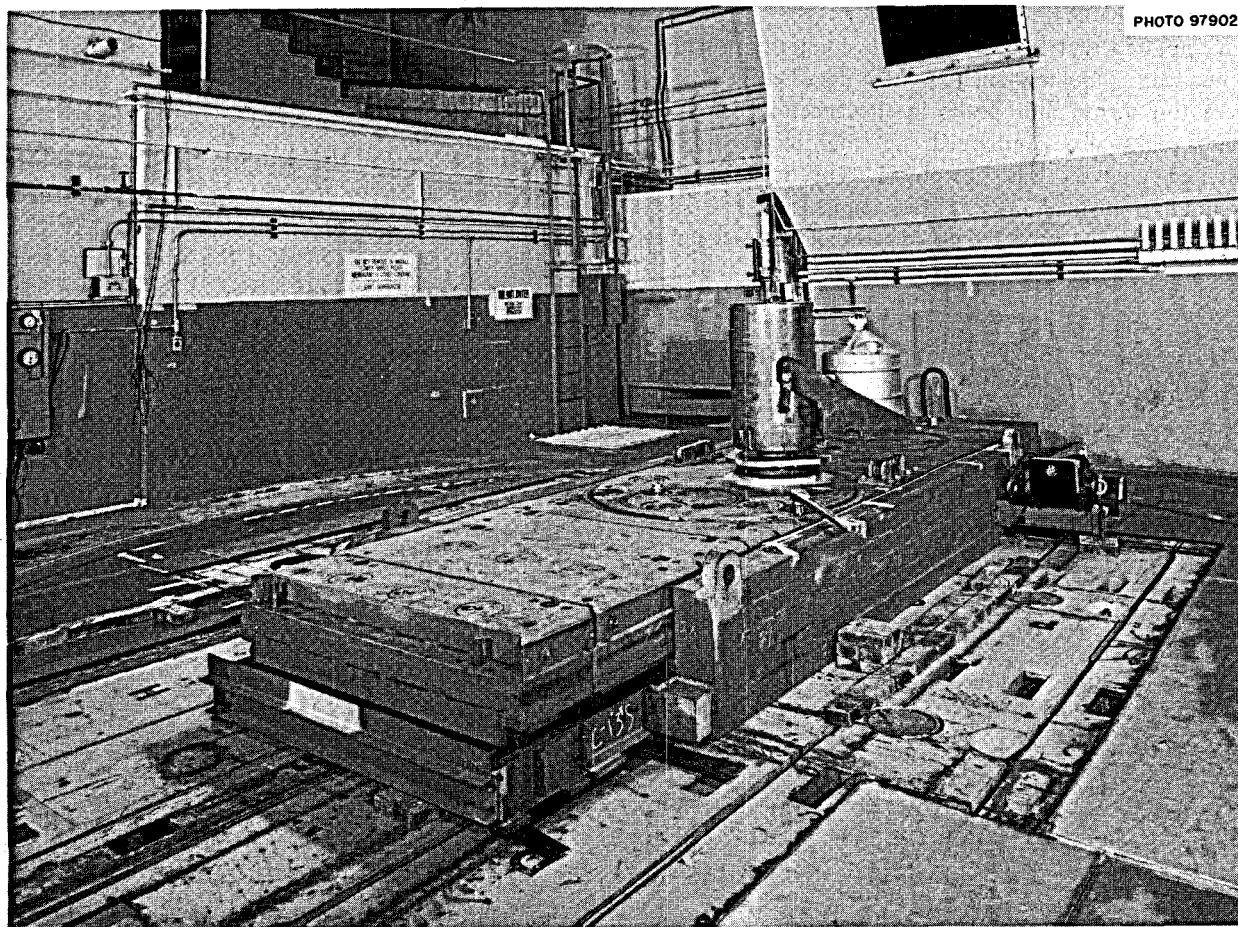


Fig. 1.8. Remote Gamma-Ray Spectrometer on Portable Maintenance Shield.

the off-gas line during different reactor conditions; xenon behavior was of primary interest. During this experiment the reactor power was varied from zero to 7.5 Mw; other variables were the fuel pump speed and the cover gas flow. Spectra were also taken when fuel salt was returned from the overflow tank to the pump bowl, during a beryllium exposure in the fuel system, and when argon was used instead of helium for the cover gas. Altogether some 170 different gamma-ray spectra were recorded at this location. Another four spectra were taken from a sample bomb in the main reactor off-gas line about 45 to 50 min downstream from the fuel pump.

The main problem encountered was the very intense activity emitted by the reactor off-gas line. When the reactor was at power, the radiation beam was more than 1000 r/hr directly above the hole in the shielding; the neutron flux was also appreciable. By using the $\frac{1}{16}$ -in.-diam collimator, 2 in. of lithium-impregnated paraffin, and a $\frac{1}{4}$ -in. lead plate, we were able to keep the counting rate down and the detector-system dead time within reasonable limits.

It appeared that the then available computer program would not adequately analyze these data because there were too many photopeaks in the spectra (many of these peaks also overlapped, i.e. multiplets), so the detailed analysis was postponed. The results of a preliminary manual analysis are given in Table 1.4. This listing shows the expected noble gases and at least some

of their daughters, but it also shows clearly the presence of noble metals.

Calibration. — An appreciable effort was devoted to calibration of the equipment. The purpose was not only to determine the counting efficiency of the equipment in the existing geometry but also to establish the self-shielding effect of the MSRE heat exchanger along with the heater elements. We used two sources for this calibration work, a small ^{226}Ra source to determine the detector sensitivity over a wide energy range and a ^{110m}Ag source of approximately 25 curies to evaluate the geometry and shielding effects. The 6.4-in.-long by 0.5-in.-diam silver source was placed in all the different tube positions of the heat exchanger mockup. This way it was possible to determine the effect on the detector of activity in each of the different tube positions. The silver source was also used to determine the detector counting efficiency for the reactor off-gas line geometry. Shielding experiments were done for the different shielding materials used (lithium-impregnated paraffin, lead, copper, aluminum, cadmium, and a heater element).

Since the source was made up of three short silver tubes that had been activated in the ORR, it was thought necessary to determine the activity along the length of the source to establish an average source strength in relation to the detector efficiency for a given collimator geometry. Local variations of 13% from the average were found in the source. However, when the normal field of view at the source was considered, the effect of these variations on count rate at the detector was much smaller.

Approximately 435 calibration and related spectra were taken, and the analysis is 95% complete. These calibration data will be used as input data for the automated computer analysis of the fission product spectra.

Scanning During Reactor Shutdowns. — The shutdown scanning program was started just prior to the reactor shutdown on November 2 and continued for almost three weeks in an around-the-clock operation. The reactor was drained from full power, and no flush salt was circulated. Gamma spectra were taken primarily from the MSRE heat exchanger and main reactor off-gas line; other spectra were taken from the fuel pump bowl, the drain tank, and two fuel salt lines (lines 101 and 102). Multiple spectra were taken to facilitate the identification of both short-lived and longer-lived species. This approach also tended to reduce statistical errors.

Table 1.4. Nuclides Identified in a Spectrum Taken from the MSRE Off-Gas Line

Reactor power, 5.5 Mw; September 1, 1969

Chain	Elements
87	Kr
88	Kr, Ru
89	Kr, Ru
90	Kr
95	Nb
99	Mo
103	Ru
106	Ru
129	Te (m)
131	Te, I
132	Te, I
133	Te(m)
135	Xe(m)
137	Xe, Cs
138	Xe, Cs
139	Xe, Cs
140	Xe, Cs, Ba, La

1.3.2 Control Rods and Drives

M. Richardson J. R. Engel

All three control rod and drive assemblies performed quite satisfactorily, with no difficulties, during this report period. Tests in late September, early October, and before the startup in November showed that scram times (times for release plus travel from full withdrawal to lower limit) were all less than 0.81 sec.

Some studies of the system used to extract control rod acceleration from records of position vs time during drop tests²⁵ were performed to explore the sensitivity of the system to changes in acceleration. In these studies an analog computer was used to generate a voltage signal in the same range as that produced by the position potentiometer on the actual rod drive. The time variation of this voltage was made to simulate rod position as a function of time for arbitrarily specified values of acceleration as a function of rod position. The simulated signals were passed through the same data recording and processing system used for the actual signals so that a direct comparison could be made between real acceleration values and those produced by the processing program from position data. Comparative results were good for long regions (more than 6 in.) of low acceleration (as was the case for the stuck control rod²⁵ in June 1969). However, if a region of zero acceleration as short as 2 in. was present at a location more than 12 in. below the starting position of the rod, the effect on the final curve of acceleration vs position was nearly indistinguishable from other random variations. The failure of the program to reproduce such small aberrations was attributed to the filtering and data smoothing that were required to reduce random noise effects.

Because of the difficulty in identifying short regions of low acceleration in a single rod drop from the fully withdrawn position, the rod testing procedure was modified to improve its fault detection capability. Accelerations were evaluated for rod drops from just above the normal operating positions as well as from fully withdrawn. This would permit detection of any tight spots within the first few inches of travel during a drop. Since a very large restriction would be required to stop a rod that has accelerated normally for 6 in. or more, these measurements, coupled with the total drop times, gave adequate assurance of rod drop capability.

Such tests were performed at least monthly during the final periods of operation and at the start of each run. There was no further evidence of abnormal drag on any of the rods.

1.3.3 Off-Gas Systems

A. I. Krakoviak

Although chronic plugging recurred in both the fuel and coolant off-gas systems during this report period, the restrictions did not interrupt power operations.

The early part of run 19 was devoted to a study of the effects of cover gas solubility on xenon stripping, which required several switchovers between helium and argon and finally back to helium as the supply cover gas. This caused perturbations in system pressure control. The inlet gas flow is normally held constant, and the fuel system pressure is controlled by manually throttling the valve at the outlet of the charcoal beds (V-557B). The transit times (assuming slug flow) from the inlet meter to the charcoal bed inlets and to the outlet throttle valve are approximately 2 and 6 hr respectively. The varying gas flow rates (due to the different physical properties of the gases) at these three primary restrictions presented somewhat of a problem of pressure control and pressure drop interpretation during the transition from one gas to another. No other operational difficulties were associated with the use of argon as a cover gas.

After approximately ten days of run 19 operations at the 5.5-Mw level, the pressure drop across the main charcoal bed increased from 3.0 psi with sections 1A and 1B in service to 4.6 psi with all three sections in service. The restrictions were partially cleared by lowering the water level in the charcoal bed pit and consecutively energizing the heaters at the entrance region of each bed for periods of 8 hr each. The pressure drop across two beds in parallel was thereby reduced to 2.7 psi; however, plugging at the beds gradually increased over the next 12 days and reached a pressure drop of 4.9 psi. The restriction then remained constant for the next 20 days, after which it was necessary to valve in section 2B also. Five days later it was necessary to lower the water level in the charcoal bed pit and reheat the inlet section of each bed. At the end of each 8-hr heating cycle, the heated bed was forward-blown with helium at 25 psig. This procedure cleared the beds so that the pressure drop across beds 1A and 1B in parallel was lowered to 2 psi. No further problems were encountered with the main charcoal beds in the subsequent run.

²⁵MSR Program Semiann. Progr. Rept. Aug. 31, 1969, ORNL-4449, pp. 16-17.

ORNL 4541
UC-23 - Reactor Technology

**CONCEPTUAL DESIGN STUDY OF A
SINGLE-FLUID MOLTEN-SALT BREEDER REACTOR**

Molten-Salt Reactor Program Staff

Compiled and edited by

Roy C. Robertson



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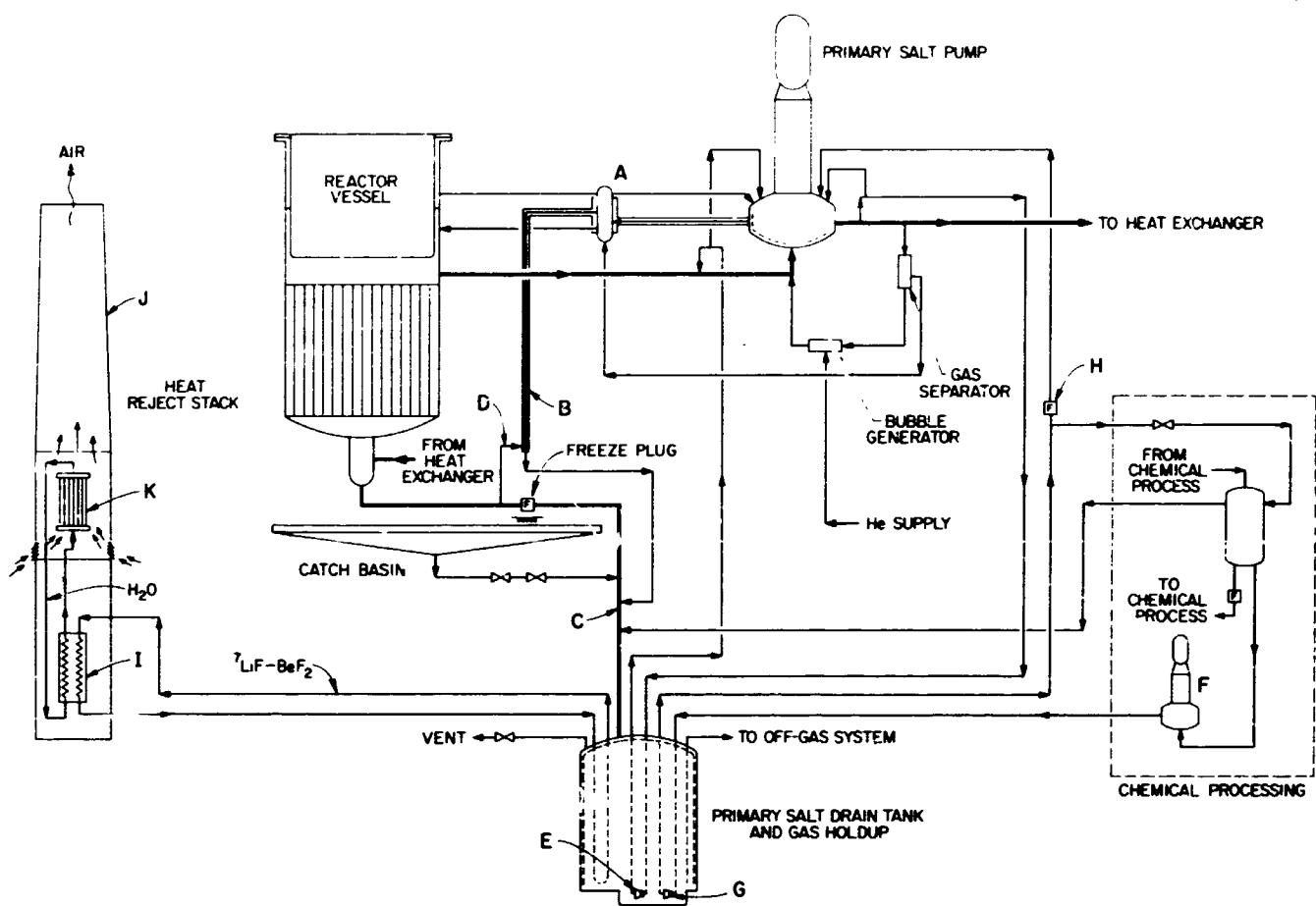


Fig. 2.3. Simplified flow diagram of primary drain tank and heat removal system using $^7\text{LiF-BeF}_2$ salt as coolant. (A) Combiner tank for separated gases and overflow salt, (B) Off-gas line with cooling jacket, (C) Fuel-salt drain line, (D) Drain line continuous bleed flow, (E) Jet pumps for returning overflow fuel salt to primary system, (F) Ancillary fuel-salt transfer pump, (G) Jet pump for filling primary system and sending salt to chemical processing, (H) Freeze-plug type valve, (I) $^7\text{LiF-BeF}_2$ -to- H_2O heat exchanger, (J) Natural draft stack, (K) Water-to-air heat exchanger.

2.5 OFF-GAS SYSTEM

The off-gases will be held in the fuel-salt drain tank for about 2 hr, during which time a portion of the noble metals will probably deposit on the internal surfaces. Referring to Fig. 2.1, the gases vented from the drain tank pass through particle traps, where remaining particulates are removed before the gases enter the charcoal beds for absorption and 47-hr holdup of the xenon, permitting decay of 97% of the ^{135}Xe . Most of the gas leaving the charcoal bed is compressed for reintroduction into the salt-circulation system at the bubble generators. A small portion of the gas leaving the 47-hr charcoal bed enters the long-delay charcoal bed (about 90-day xenon holdup), the outflow of which passes through tritium and krypton traps before entering a gas storage tank. The gas from this tank is augmented by makeup helium if required and reintroduced into the circulation system as purge gas for the circulation pumps and at other places where clean helium is needed. The accumulated krypton and tritium are stored in tanks in the waste cell facility.

duced into the circulation system as purge gas for the circulation pumps and at other places where clean helium is needed. The accumulated krypton and tritium are stored in tanks in the waste cell facility.

2.6 FUEL-SALT PROCESSING SYSTEM

L. E. McNeese

Breeding with thermal neutrons is economically feasible with a molten-salt reactor because it is possible to process the fluid fuel rapidly enough to keep the neutron losses to protactinium and fission products to a very low level. The equipment used to strip gaseous fission products from the fuel salt was described in Sects. 2.1 and 2.5. The concentrations of protactinium, rare earths, and some other fission products are limited by continuously processing a small stream of the fuel salt in an on-site processing system, described below.