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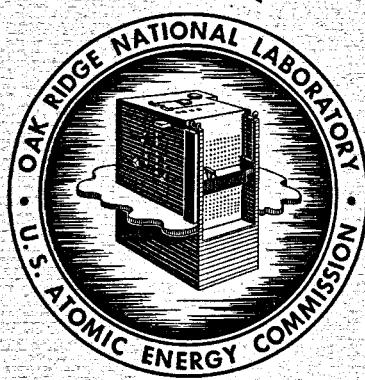
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ORNL-3791
UC-80 - Reactor Technology
TID-4500 (46th ed.)

PRELIMINARY DESIGN STUDY OF A CONTINUOUS
FLUORINATION-VACUUM-DISTILLATION
SYSTEM FOR REGENERATING FUEL AND
FERTILE STREAMS IN A MOLTEN
SALT BREEDER REACTOR

C. D. Scott
W. L. Carter

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS



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

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ORNL-3791

Contract No. W-7405-eng-26

CHEMICAL TECHNOLOGY DIVISION

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PRELIMINARY DESIGN STUDY OF A CONTINUOUS FLUORINATION-
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ABSTRACT

The purpose of this study was to make a preliminary design and an engineering evaluation of a conceptual plant for treating the fuel and fertile streams of a molten-salt breeder reactor. The primary requirements of the process are to recover the unburned fuel ($^{233}\text{UF}_4$) and fuel-stream carrier salts (LiF-BeF_2) from the fuel stream, and the LiF-ThF_4 plus the bred ^{233}U from the fertile stream. Both streams must be sufficiently decontaminated for attractive breeding performance of the reactor. The plant was designed to operate continuously as an integral part of the reactor system, fitting into two relatively small cells adjacent to the reactor cell. In this study, the plant capacity is based upon treating 15 ft³/day of fuel salt and 105 ft³/day of fertile salt removed as side streams. These capacities are adequate for a 1000-Mw (electrical) power reactor.

As to the fuel stream, basically it is purified by fluorination and vacuum distillation. The first step removes uranium as volatile UF_6 ; the second recovers the LiF-BeF_2 by simultaneously volatilizing these two components from the less volatile fission products. Fortunately, the fission products so separated are primarily the rare earths, which are the most serious neutron poisons. The UF_6 from the fluorinator is accompanied by some volatile fission product fluorides, primarily those of Mo, Te, Ru, Zr, and Nb, which are removed by sorption on granular NaF and MgF_2 . Finally, the UF_6 is reduced to UF_4 by hydrogen and recombined with the decontaminated LiF-BeF_2 carrier in a single operation. Fission products are removed from the plant by discard of NaF and MgF_2 sorbents and the still residue, which is a highly concentrated solution of the rare earths in LiF. Wastes are permanently stored underground.

With respect to the fertile stream, the process consists only of fluorination followed by decontamination of the UF_6 on NaF and MgF_2 sorbents. It is only necessary to remove the bred ^{233}U sufficiently fast to keep a low concentration in the blanket, thereby ensuring a low fission rate and negligible

poisoning by fission products. Discard of the barren fertile stream at a slow rate suffices to keep the fission product concentration at a tolerable level.

The chief conclusions of this study are: (1) that the fluorination-distillation process for the fuel stream and the fluorination process for the fertile stream comprise a compact and relatively simple system that can be engineered with a normal amount of developmental work, and (2) that integration of the processing plant into the reactor facility is both feasible and economical and the logical way to take advantage of the processing possibilities of a fluid-fueled reactor. The nominal cost of this plant is presented in the following summary of major items:

Process equipment and building space	\$5,302,000
Fuel salt inventory	89,500
Fertile salt inventory	69,200
NaK coolant inventory	40,000
Direct operating cost	788,000/year

These costs contribute about 0.2 mill/kwhr to the fuel cycle cost when the reactor operates at an 80% plant factor and capital charges are amortized at 10%/year. This cost is sufficiently low to add to the incentive for developing the molten-salt breeder reactor.

Some of the steps of the evaluated process are based on well-established technology, whereas others are based on extrapolations of laboratory and small-scale engineering data. Fluoride volatility and associated UF₆ decontamination by sorption are well-known operations, having been demonstrated in a pilot plant. However, vacuum distillation and liquid-phase reduction of UF₆ to UF₄ have been demonstrated only at the bench. Certainly, more development of these steps is required for a complete process demonstration. A singularly serious problem is the corrosive nature of the fluorine-molten salt mixture in the fluorinator. However, this study shows that this and other inherent processing problems can be solved by proper design and operation of equipment.

INTRODUCTION

A reactor concept that has a high potential for economic production of nuclear power and simultaneous breeding of fissile material on the thorium-uranium fuel cycle is the molten-salt breeder reactor (MSBR). This reactor utilizes two fluid streams. For this study, the stream compositions are: (1) a fuel stream consisting of an LiF-BeF₂ (69-31 mole %) carrier that contains the fissile component, and (2) a fertile stream, which surrounds the fuel stream, consisting of a 71-29 mole % LiF-ThF₄ mixture. In each stream the lithium is about 99.995 at. % ⁷Li. The above compositions are not unique; other MSBR designs might use different compositions.³⁹ The configuration of the system allows a relatively high neutron leakage rate from the fuel stream into the surrounding fertile stream, where capture by thorium breeds additional ²³³U fuel. The reactor is operated at an average temperature of about 650°C; fission energy is recovered in external heat exchangers through which the fuel and fertile streams are circulated. Sustained breeding performance of the MSBR depends on the removal of fission and corrosion products from the two fluid streams so that parasitic neutron capture is kept to a tolerable rate. Portions of the two circulating streams are continuously removed, processed for removal of fission products, fortified with makeup fissile and fertile materials, and returned to the reactor in a cyclic operation.

A primary consideration of any process for recycling reactor fuel is that minimal losses of all valuable fuel components be attained without intolerable capital investment or operating expense. In the MSBR system, this requirement applies to both fuel and carrier components. As mentioned above, the fuel is ²³³U, and one carrier component is the highly enriched ⁷Li isotope. The other major constituents, beryllium and thorium, are less expensive; yet large losses of these cannot be tolerated either because of the adverse effect on fuel-cycle cost and fuel conservation. The process evaluated here accomplishes the objectives of conservation while providing fission-product removal sufficient for a successful breeding system.

The work reported here is unique in that it examines a processing plant integrated directly into the reactor system, which, in effect, accomplishes on-stream processing. This method obviates the cumbersome

and expensive transfer of highly radioactive material by carrier shipment; furthermore, common use can be made of services and equipment necessary to the reactor, thus avoiding the duplication that results in a separate processing building. Also, since the spent fuel flows directly into the processing plant, there is minimal out-of-pile inventory of valuable fuel components.

Another interesting feature of this study is the use of the relatively recent concept of vacuum distillation as a means of purifying the carrier salt. A modest vacuum of only 1 to 2 mm Hg is required, but the temperature (about 1000°C) is higher than any normally encountered in handling molten fluoride salts. The operation was explored first by Kelley³ in laboratory experiments that indicated LiF-BeF₂ decontamination from fission products by factors of 10² to 10³. The attractiveness of the process lies in the fact that it involves only a physical operation that is easily controlled and that can be made continuous. Fission products can be concentrated in the still residue (primarily ⁷LiF) by a factor of about 250 by using the decay heat of the fission products to volatilize LiF-BeF₂. This high concentration factor ensures a low discard rate for the valuable ⁷Li. Cyclic operation of the still was assumed, allowing the fission product concentration to increase with time. The corresponding increase in the rate of decay heat generation limited the cycle time to about 68 days. Although limited experience with the distillation step indicates that high-nickel alloys are satisfactory structural materials for use at this relatively high operating temperature, a more extensive investigation is needed to define the design limitations.

A novel idea has also been studied in the evaluation of liquid-phase reduction of UF₆ to UF₄ by hydrogen. Initial bench-scale experiments²⁶ have given promising data. The reaction is carried out by absorbing UF₆ in a molten mixture of LiF-BeF₂-UF₄ at about 600°C followed by contacting with H₂ to reduce UF₆ in situ. This technique avoids the troublesome problem of remotely handling solids (small UF₄ particles) that would be met if the customary gas-phase reduction of UF₆ were used.

Aside from indicating the feasibility and economy of the process, this study also uncovered important design and engineering problems associated with the scaleup of laboratory and batchwise operations to

larger, continuous ones. In this regard, recommendations are presented at the end of the text, along with important conclusions. The most noteworthy recommendation is that the key operations, vacuum distillation and continuous fluorination, be given high priority in development.

The material that follows is arranged in this sequence: First, a brief description of the MSBR system is given to put the study in perspective; second, design criteria and ground rules are stated for each phase of the study; third, a process flowsheet and a description of each unit operation is presented; fourth and fifth, a description and pertinent design data for each major component and the processing cells are listed; sixth, equipment and building-cost data are presented; seventh, an overall evaluation of the process is given in a set of enumerated conclusions and recommendations arranged according to plant characteristics and process operations. Six Appendices, giving detailed data and calculations, are attached.

THE MOLTEN-SALT BREEDER REACTOR SYSTEM

The processing system of this study is designed to meet the requirements of the molten-salt breeder reactor shown in Fig. 1. This is a conceptual design¹ of a power reactor capable of producing 1000 Mw (electrical) with a thermal efficiency of 45%. Basically, it consists of a graphite matrix enclosed in a cylindrical Hastelloy N vessel for containment. Graphite occupies about 79 vol % of the core, fuel salt about 15 vol %, and fertile salt about 6. The flow passages are such that the fuel and fertile streams do not mix. The core is surrounded radially and axially by a 3.5-ft blanket of LiF-ThF₄ mixture, and the blanket is in turn surrounded by a 6-in.-thick graphite reflector. The core is about 8 ft in diameter and 17 ft high; overall, the reactor plus breeder blanket is about 16 ft in diameter and 25 ft high.

Fission energy is recovered in a battery of external heat exchangers through which the fuel and fertile streams are continuously circulated. The coolant may be either a molten carbonate or fluoride salt mixture which transports the heat to boilers for producing steam. Small side-streams of fuel and fertile fluids are continuously withdrawn from the

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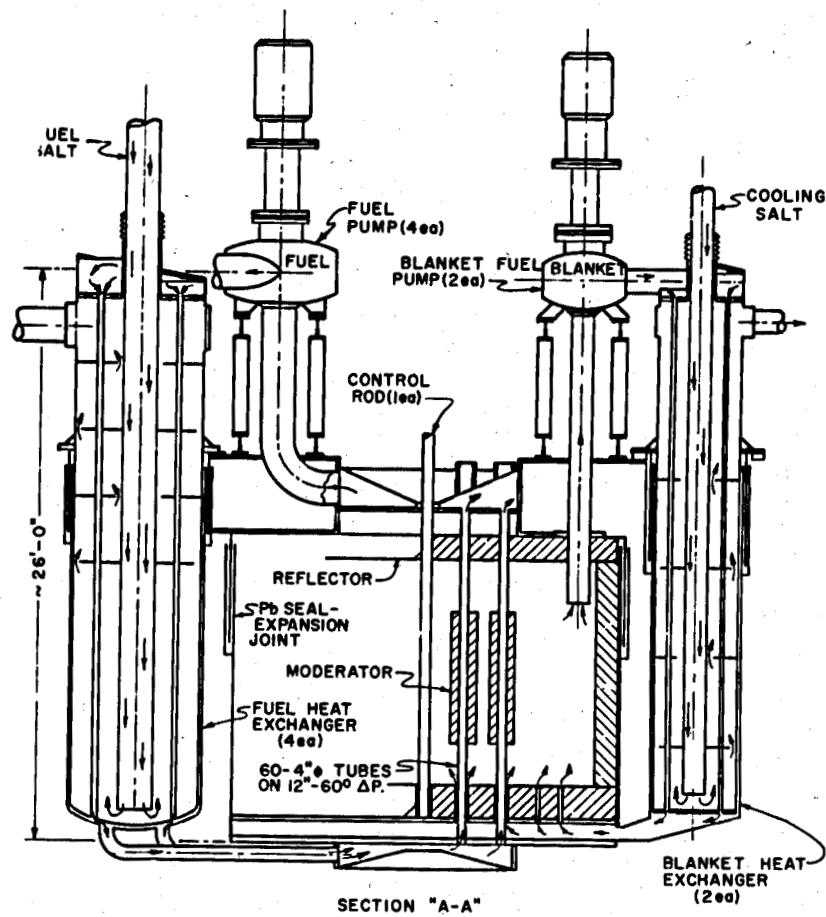
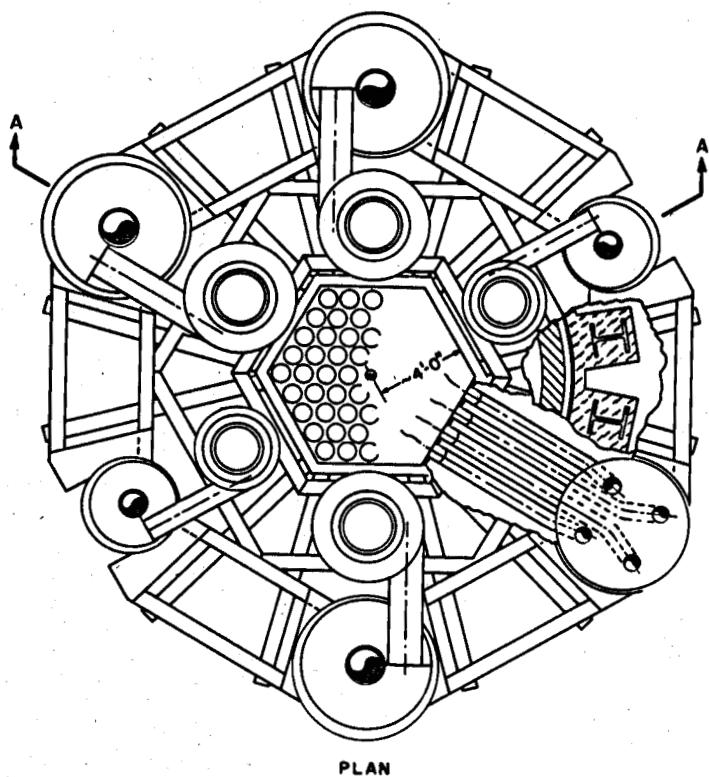


Fig. 1. Conceptual Design of the Molten-Salt Breeder Reactor.

circulation loops and routed to a chemical processing cell adjacent to the reactor cell. After being processed for fission product removal and reconstituted with makeup materials, the two fluids are returned to the reactor via the fuel-makeup system. The processing cycle is selected to give the optimum combination of fuel-cycle cost and breeding gain.

The data presented in Table 1 are typical for the MSBR and were the bases for this study.² Since the reactor concept is undergoing engineering and physics evaluation, these data represent no fixed design and are subject to change as the studies progress.

DESIGN CRITERIA

The following discussion delineates ground rules and arguments for the particular choice of process and design used in this evaluation. Choices were made on the basis of existing knowledge and data. The study presented here is expected to verify basic assumptions or indicate judicious alternatives.

Basic Considerations

One basic consideration concerns the fuel yield (the fraction of fissile inventory bred per year) which, for a breeder reactor, is inversely proportional to the total inventory of the reactor and chemical plant systems. This characteristic is essential to the design of an MSBR processing plant and suggests close-coupling of the reactor and processing plant to give minimal out-of-reactor inventory. A fluid-fueled system is readily amenable to this type of operation, and for this evaluation the processing plant is integrated with the reactor plant. This design permits fast, continuous processing, restricted only by the rather stringent design requirements for fission-product decay-heat removal and corrosion resistance.

The integrated plant occupies cells adjacent to the reactor cell, and all services available to the reactor are available to the chemical plant. These include mechanical equipment, compressed gases, heating and ventilating equipment, electricity, etc. The cost savings for an integrated

Table 1. Typical Characteristics of a Molten Salt Breeder Reactor

General	
Reactor power, Mw (electrical)	1000
Thermodynamic efficiency, %	45
Reactor geometry	Cylinder
Core diameter, ft	8
Core height, ft	17
Blanket thickness, ft	3.5
Moderator	Graphite
Volume fraction of moderator in core	0.79
Volume fraction of fuel in core	0.15
Volume fraction of fertile stream in core	0.06
Reactor containment vessel	INOR-8
Fraction of fissions in fuel stream	0.972
Plant factor	0.8
Breeding ratio	~1.08
Fuel Stream	
Composition, mole %	
LiF (99.995 at. % ^{7}Li)	68.5 ^a
BeF ₂	31.2 ^a
UF ₄ (fissile)	0.31
Inventory at equilibrium	
Volume, ft ³	671
LiF, kg	19,530
BeF ₂ , kg	16,100
^{233}U (as UF ₄), kg	736
^{235}U (as UF ₄), kg	11.1
Other U (as UF ₄), kg	58.1
Cycle time, days	58
Power, Mw (thermal)	2160
Liquidus temperature, $^{\circ}\text{C}$	~500
Density (calculated)	
ρ (g/cm ³) = $2.191 - 0.0004 t$ ($^{\circ}\text{C}$) for $525 \leq t \leq 1200$ $^{\circ}\text{C}$	
Tertile Stream	
Composition, mole %	
LiF (99.995 at. % ^{7}Li)	71
ThF ₄	29
$^{233}\text{UF}_4$	0.012 ^b
$^{233}\text{PaF}_4$	0.022 ^b
Inventory at equilibrium	
Volume, ft ³	1783
LiF, kg	38,760
Th (as ThF ₄), kg	141,290
^{233}U (as UF ₄), kg	60
^{233}Pa (as PaF ₄), kg	110
Cycle time, days	22
Power, Mw (thermal)	62
Liquidus temperature, $^{\circ}\text{C}$	~565
Density (calculated)	
ρ (g/cm ³) = $4.993 - 0.000775 t$ ($^{\circ}\text{C}$) for $565 \leq t \leq 1200$ $^{\circ}\text{C}$	
Steam Conditions	
Pressure, psia	3515
Temperature, $^{\circ}\text{F}$	1000
Condenser pressure, in. Hg abs	1.5

^a Basic composition of carrier salt is 69.31 mole % (LiF-BeF₂).

^b Equilibrium composition for this cycle.

facility are immediately apparent when one considers the large amount of equipment and facility duplication required for separate plants.

A further basic consideration is that there will be no large extrapolation of technology in the process design. Accordingly, the process is based on treating the molten salt by fluorination and distillation, with the supporting operations of UF_6 sorption on and desorption from beds of pelletized NaF , followed by reduction of the UF_6 to UF_4 . A large amount of data is available for each step except for the distillation and reduction operations, both of which have been demonstrated in the laboratory.^{3,26} However, this study does assume the necessary engineering extrapolations to convert from the current batchwise operations to continuous operations.

Process for the Fertile Stream

The two streams of the breeder reactor require different processing rates and must be treated separately to prevent cross contamination. The first step in process for the fertile stream consists only of continuous fluorination, which removes the bred uranium as the volatile hexafluoride. No other treatment is needed if this step is designed to maintain a low uranium concentration. To accomplish this, the stream is required to go through the processing plant on a relatively short cycle, for example, once every 20 to 50 days. The cycle time for this study is 22 days. The fission rate in the blanket is low, and the fission products are kept at a tolerable level by periodic discard of barren $\text{LiF}-\text{ThF}_4$ salt. A 30-year discard cycle suffices. In the second step, the volatilized UF_6 is sorbed on NaF beds, desorbed, and finally caught in cold traps.

Process for the Fuel Stream

The fuel stream of the reactor is processed by fluorination and vacuum distillation to recover both uranium and carrier salt sufficiently decontaminated of fission products. A cycle time of 40 to 70 days is required to maintain the fission-product concentration at a low enough level for attractive breeding performance. The calculations of this study

are for a 58-day cycle. The UF_6 is recovered by NaF sorption and cold traps, just as for the fertile stream. Decontaminated fuel and carrier are recombined in a reduction step that converts UF_6 to UF_4 and further purifies by reducing corrosion products (iron, nickel, chromium) to their metallic states. Makeup fuel and carrier are added at this point, and the stream is returned to the reactor.

The time that the fuel stream spends in the processing plant is kept as short as practicable to minimize out-of-reactor inventory.

Waste Storage

The chemical plant provides its own storage system for process wastes. Incidental wastes, such as slightly contaminated aqueous solutions and flush salts, are assumed to be handled by the reactor waste system, thus such facilities are not duplicated for the processing plant. Separate storage is provided for fuel-and fertile-stream wastes, which are primarily LiF plus fission products, and LiF- ThF_4 , respectively. The facilities are designed for a 30-year capacity and are located underground a short distance from the chemical processing area.

The fuel-stream process also produces a less radioactive waste than the LiF-fission-product mixture. This waste is in the form of pellets of sodium fluoride and magnesium fluoride pellets used for decontaminating uranium hexa-fluoride. Interim storage of 5-year duration is provided for these solids.

Fission-product decay heat is removed either by forced air or natural convection, as required by the heat load.

Operating Policy

Certain ground rules consistent with convenient and safe operation were adopted for this study. Maintenance operations are facilitated by assigning unit-process steps to either a high- or a low-radiation level cell. Operating and maintenance personnel, who are not required on a full-time basis, are to be shared with reactor operation. No water or aqueous solutions are to be admitted to the process cells; fluids required for heat transport will be either air or sodium-potassium eutectic (NaK). A barren

fluoride salt, for example, an NaF-KF mixture, would be an acceptable substitute.

Process Data

The primary concern of a processing cycle for short-cooled fluoride mixtures is in dissipating fission product decay energy so that process operations can be controlled. A maximum of about 6.5% of the total energy of the system is associated with beta and gamma energy in the fission products; this amounts to about 140 Mw (thermal) in the MSBR fuel stream. Most of this energy is emitted quickly, decreasing about 82% in 1 hr and 95% in 1 day. The data for this study were calculated for the reactor described in Table 1, using the PHOEBE Code,⁴ which computes gross fission-product decay energy as a function of exposure and time after discharge from the reactor. The data are presented graphically in Figs. 2a-b and 3a-b for fuel and fertile streams, respectively.

The graphs give an upper limit for heat generation because the calculations do not account for possible intermittent reactor operation attributable to the 80% plant factor. In addition, the graphs include decay energy associated with gaseous products that are sparged in the reactor circulating loop and with those fission products that might deposit on surfaces throughout the system. It is difficult to separate this energy from gross energy until more is known about the behavior of fission products in molten fluorides.

A process flowsheet showing material balances for the fuel and fertile streams in the processing plant is included in Appendix F.

DESCRIPTION OF PROCESS

The processing facility must have the capability of removing the major portion of fission products from the molten fuel salt and returning the purified salt to the fuel system after necessary reconstitution with ^{233}U and carrier salts. As to blanket-salt processing, the facility must achieve recovery of the major portion of the bred uranium for recycle to the fuel stream or for sale. These goals can be met with present technology or with processes that can be reasonably extrapolated from current technology.

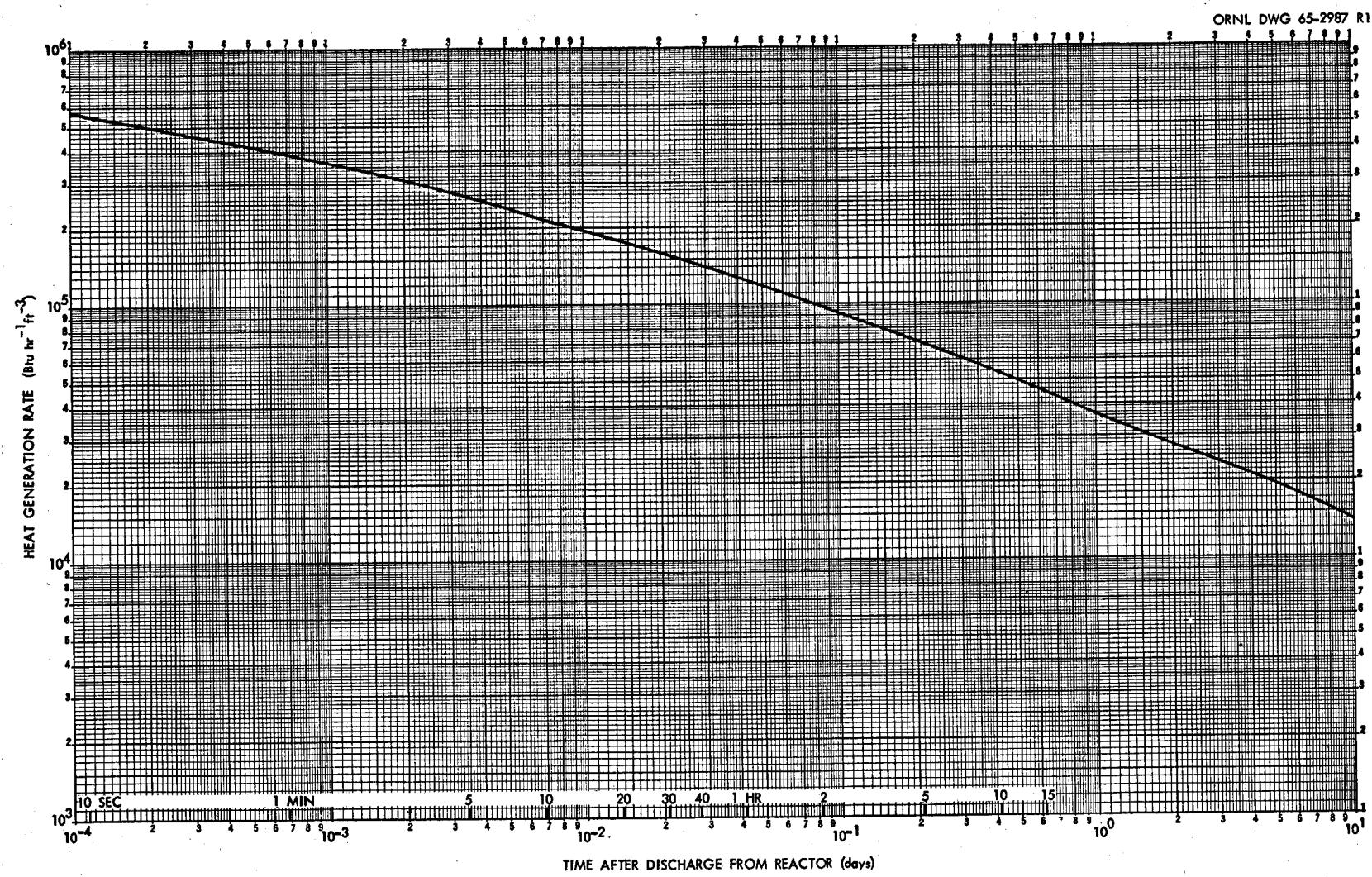


Fig. 2a. Fission-Product Decay Heat in MSBR Fuel Stream. Fuel-stream volume, 671 ft³; power, 2160 Mw (thermal); cycle time, 58 days.

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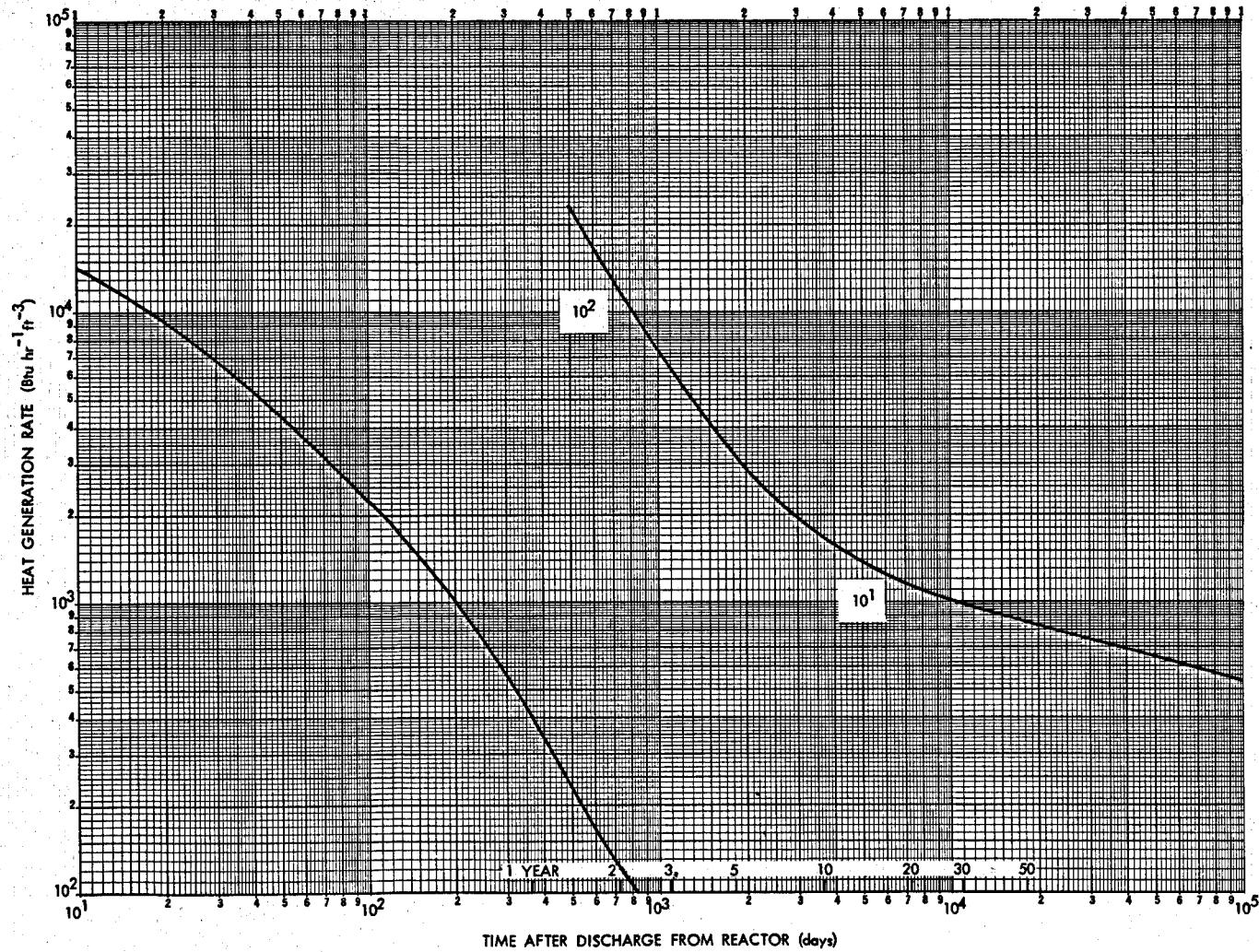


Fig. 2b. Fission-Product Decay Heat in MSBR Fuel Stream. Fuel-stream volume, 671 ft³; power, 2160 Mw (thermal); cycle time, 58 days.

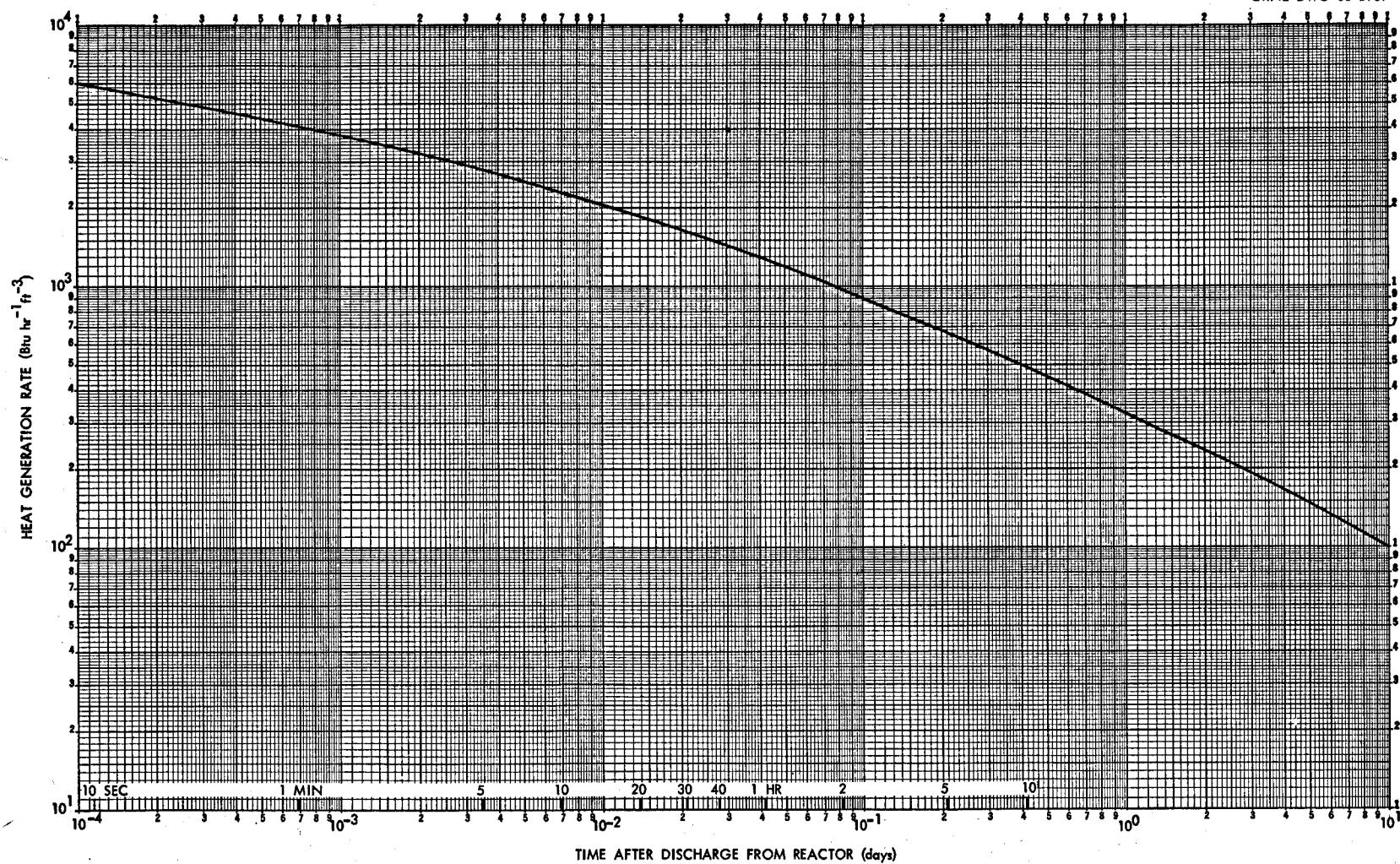


Fig. 3a. Fission-Product Decay Heat in MSBR Fertile Stream. Fertile-stream volume, 1783 ft³; power, 62 Mw (thermal); cycle time, 22 days.

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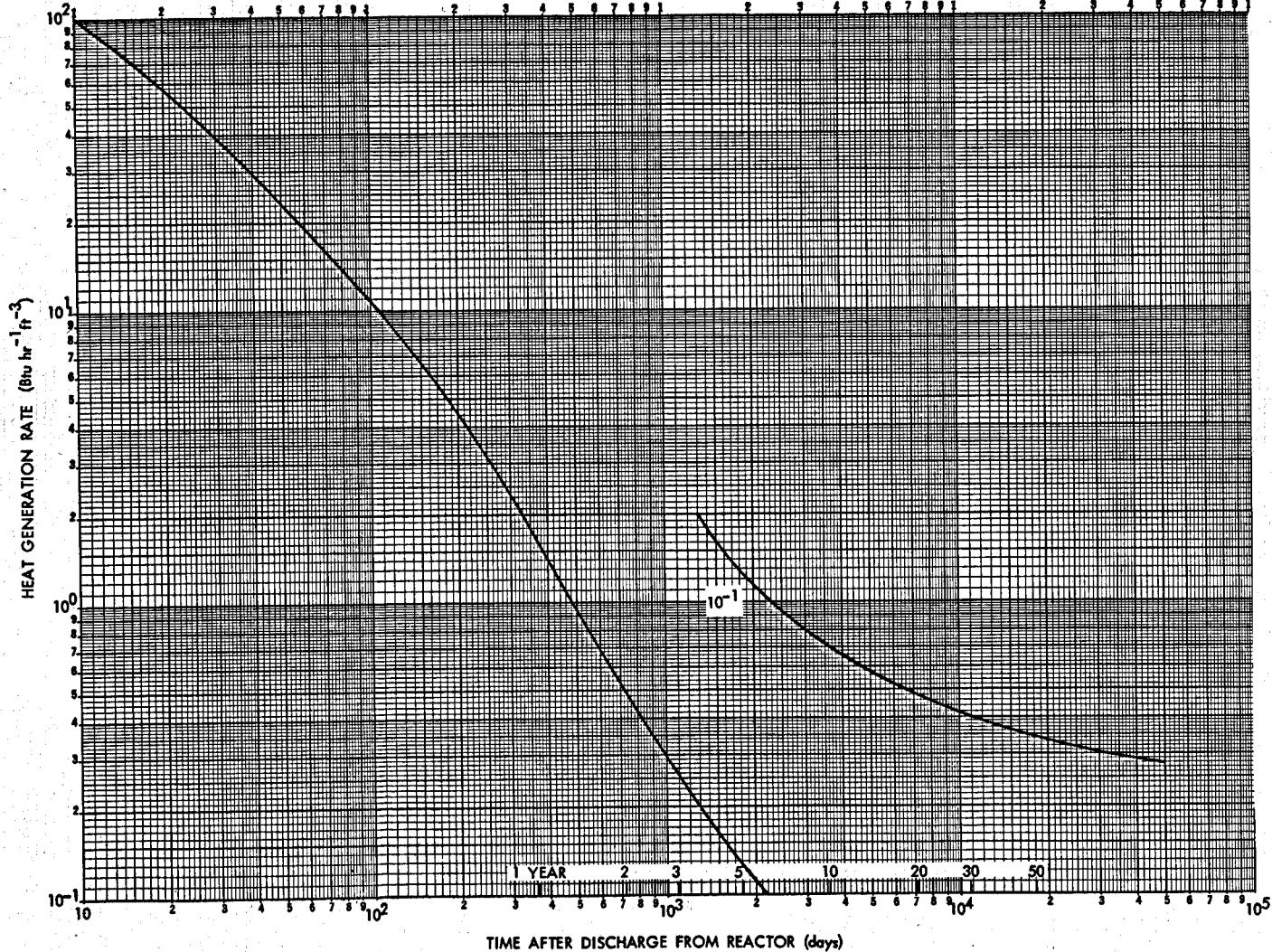


Fig. 3b. Fission-Product Decay Heat in MSBR Fertile Stream. Fertile-stream volume, 1783 ft³; power, 62 Mw (thermal); cycle time, 22 days.

Summary of the Process Flowsheet

In Fig. 4 it can be seen that fuel-stream processing is achieved by first removing the uranium (as $^{233}\text{UF}_6$) and volatile fission products from a side-stream of the molten salt by fluorination. Then the carrier salt is vacuum-distilled from the residual fission products. Next, the UF_6 , further purified by a sorption-desorption process based on the use of NaF pellets, is dissolved in the carrier salt. Finally, the UF_6 is reduced to UF_4 by hydrogen, thereby reconstituting the fuel salt.

Blanket salt is processed concurrently with the fuel salt by fluorinating a side-stream. The UF_6 gas is separated from the volatile fission-product fluorides by sorption and desorption, using beds of NaF pellets as mentioned above. The blanket salt needs no further purification, and a portion of the $^{233}\text{UF}_6$ is sent to the fuel stream by dissolving it in the carrier salt and reducing it with hydrogen. The excess over that needed to refuel the reactor is sold.

The two chemical reactions (fluorination and reduction) in the process are simple, fast, and quantitative. The other interactions are physical and require only heat and mass transport; however, in the case of distillation, a rather high temperature (about 1000°C) is involved. The salt is extremely stable at any temperature anticipated in this process, and other physical properties, primarily vapor pressure and solubility for fission products, are in accord with process requirements.

Fluorination

As noted above, uranium and volatile fission products are removed from both streams by fluorination. A batchwise, molten-salt fluoride-volatility process for recovering uranium has been under development for several years. Currently the process is in the pilot plant phase at ORNL.^{17,18} One of the steps in this process is batch fluorination of the molten fluorides to remove uranium hexafluoride. As noted before, uranium in these streams is in the form of UF_4 . This uranium and some of the fission products are converted by fluorine to higher-valent, volatile fluorides which leave the salt and go to the uranium hexafluoride

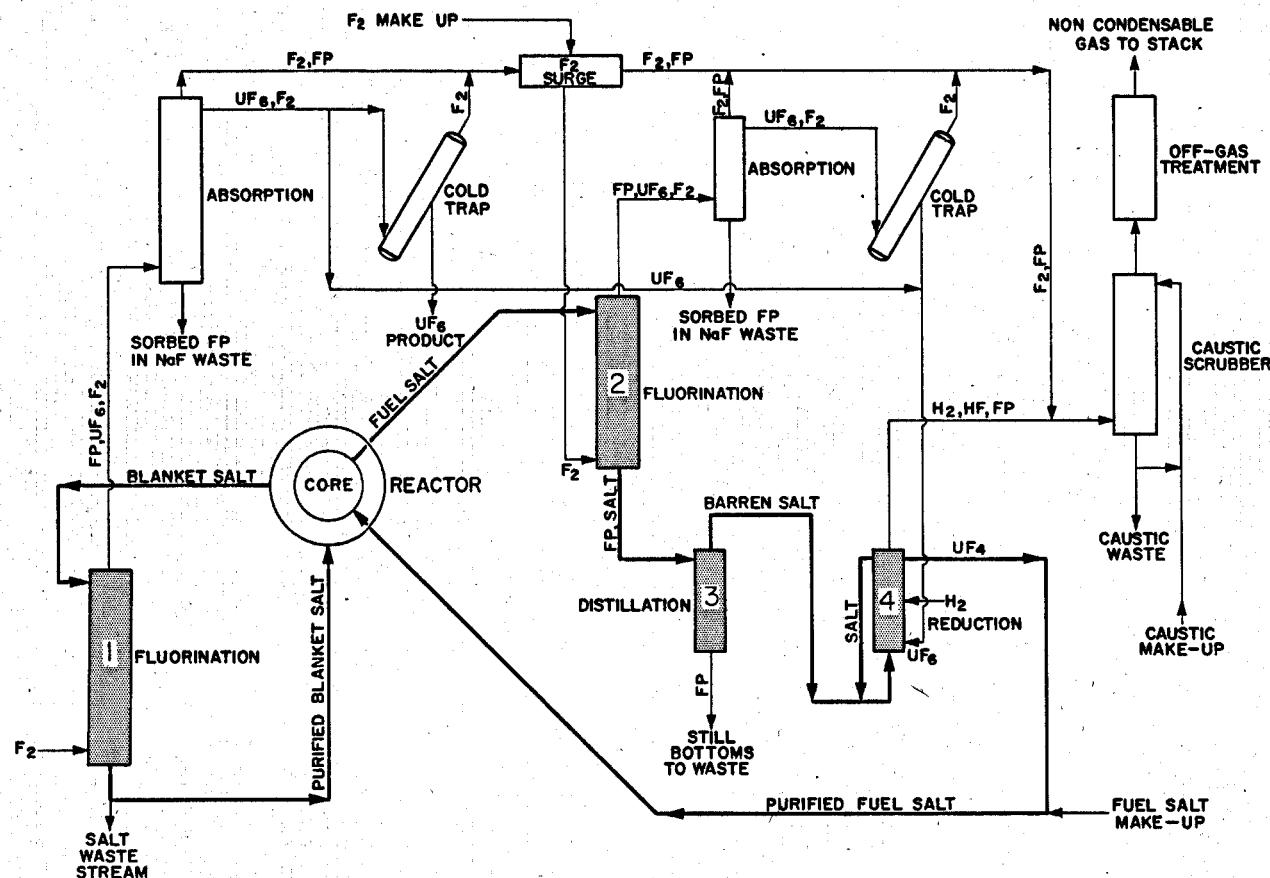
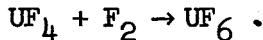


Fig. 4. Major Steps in MSBR Fuel and Blanket Processing. Uranium is separated from blanket and fuel salts by fluorination (blocks 1 and 2), further purified by sorption, and collected in cold traps. The barren fuel-carrier salt is distilled away from the bulk of the fission products (block 3). The cold-trapped UF_6 and fuel-carrier salt are recombined in reduction step (block 4) and returned to the core.

purification system. The reaction for producing UF_6 in the present batch processing is the same as that for the proposed continuous system and may be represented as follows:



Certain fission-product fluorides are also volatile. The principal ones are: Ru, Nb, Cs, Mo, and Te. Zirconium is volatilized to a lesser degree.

Continuous processing of the MSBR fuel salt can be best achieved if the fluorination process is made continuous, preferably with countercurrent contacting of the molten salt and fluorine. Such a process has not been developed, and the present volatility process at ORNL is entirely a batch process. Therefore, development work will be necessary to provide for continuous fluorination.

Rate data for molten-salt fluorination are limited and conflicting, although the reported rates have been sufficient for the batch process. Mailen's data¹⁹ on the fluorination of falling droplets of molten salt support the view that the reaction itself is very fast; whereas, the slower rates that result from sparging a pool of molten salt with fluorine (Cathers *et al.*,²⁰ Pitt,²¹ and Moncrief²²) can be explained by assuming a mass-transfer-controlling rate mechanism due to inefficient mixing of the gas and liquid phases. Contact times (fluorine and molten salt) of 1 to 2 hr have been shown to be adequate for uranium removal down to 10 ppm in such batch tests. The countercurrent, continuous operation, envisioned here, would probably give better contact. It is difficult to predict the fluorine utilization in a continuous fluorinator; however, it should be better than that for batch fluorination, which suggests that a utilization of 33-1/3% will probably be achieved.

A relatively low mass flow rate of gas must be maintained in the fluorinator to prevent salt entrainment in the off-gas. The highest mass flow rate that has been used successfully without entrainment is 0.28 slpm per in.² of fluorinator cross section.²³

Experience in the Volatility Pilot Plant at ORNL showed that corrosion on bare metal walls (L-nickel) in the fluorinator is relatively high. Kessie *et al.*²⁴ developed a technique for keeping a frozen

protective layer of salt on the metal wall, and we adopted their approach for this study. This frozen wall of salt, which is kept so by a high cooling rate on the metal wall, prevents gross corrosion. Internal heating keeps the rest of the salt molten. For the fuel stream, fission-product heat is sufficient for this purpose, but for the blanket stream it may be necessary to supplement decay heat by using suspended electrodes for resistance heating. Since a maximum of 1.5 kw of heat per ft² of wall area can be removed from such a system, the fuel stream will have to be delayed in a cooling tank until the heat generation due to decay heat is at a sufficiently low level. As to the blanket salt, decay heating will be insignificant.

The continuous fluorinator is a tall column into which molten salt is fed at the top and flows to the bottom; fluorine is introduced at the bottom and passes to the top, accompanied by the volatile fluorides (Fig. 5). An expanded deentrainment section is added to the top, and the body of the column is jacketed with a coolant to maintain the frozen salt wall. A gravity leg is used in the molten salt outlet to hold a constant salt level in the column.

Purification of Uranium Hexafluoride by Sorption and Desorption

The UF₆ from both streams is purified in the same way before being returned to the fuel stream. Since the UF₆ that leaves the fluorinator contains volatile fission products, it is purified by a series of sorption steps. These are batch steps, but the process is made continuous by using parallel beds alternately.

The first separation occurs in a NaF absorption system where the gas stream passes through fixed beds of NaF pellets. This system consists of two distinct zones, one held at 400°C and one at about 100°C. In the higher-temperature zone, corrosion products, entrained salt, and niobium and zirconium are irreversibly removed from the fluorinator off-gas while the UF₆ and other fission products pass through (Fig. 6). In the second NaF zone (100°C), UF₆ and some of the molybdenum are held up by sorption while the remaining fission products pass through. At this point, the UF₆ has been separated from all the fission products except molybdenum.

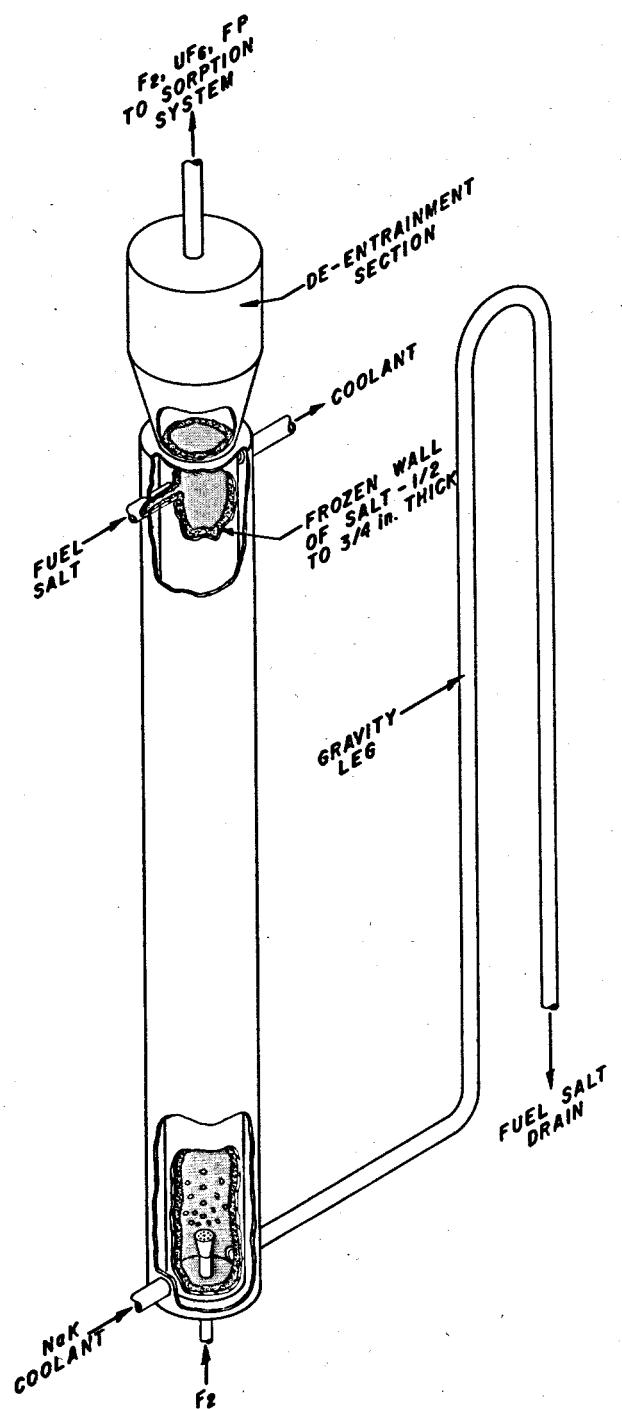


Fig. 5. Continuous Fluorination. NaK coolant, flowing through the jacket, freezes a layer of salt on the inner surface of the column, thus protecting the Alloy 79-4 from corrosive attack by the molten-salt-fluorine mixture.

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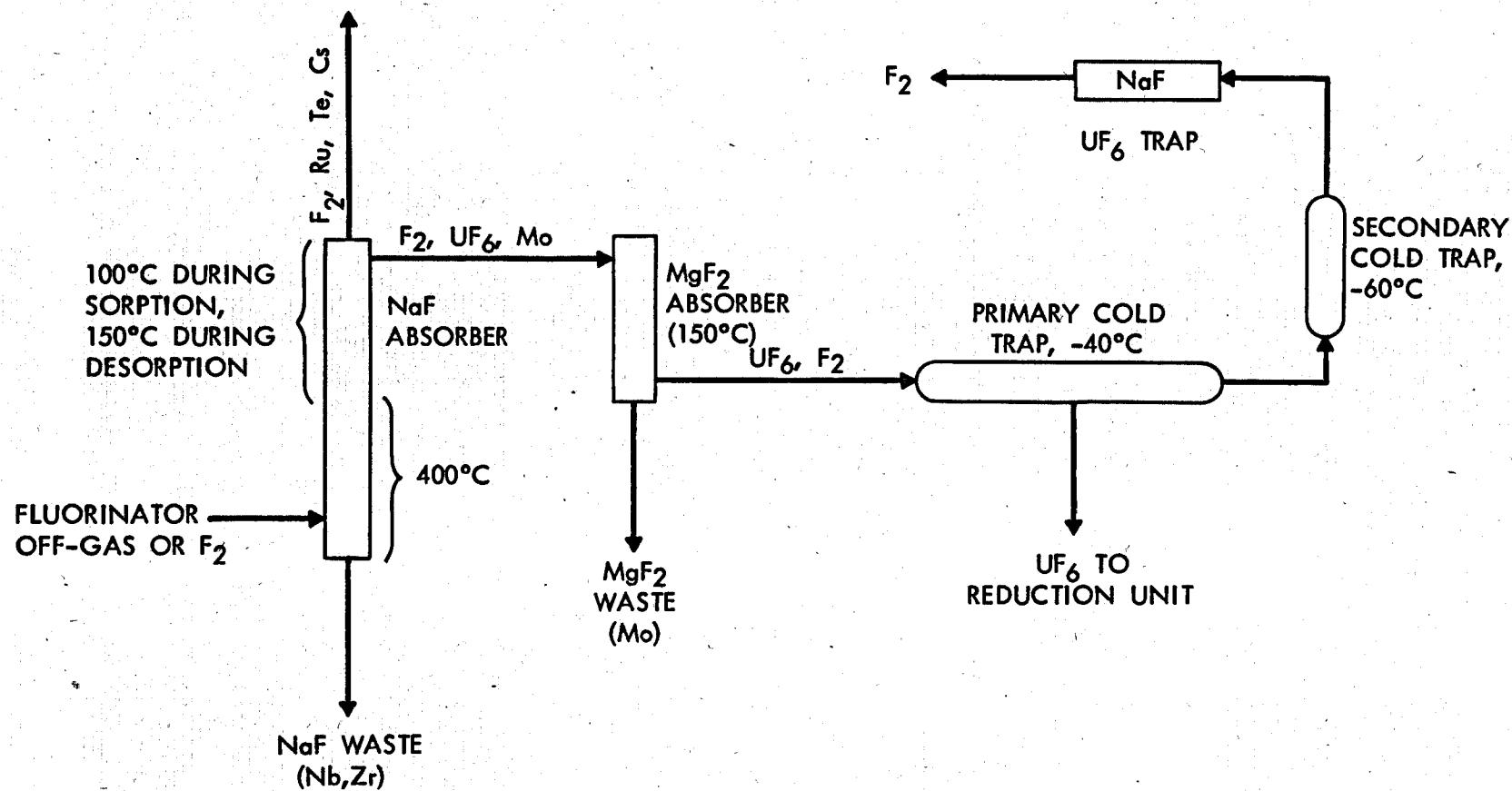


Fig. 6. Purification System for UF_6 with Disposition of Volatile Fission Products. Two such systems are used, one for the fuel stream and one for the fertile.

Molybdenum fluoride is removed from the UF_6 by isolating the 100°C NaF zone, desorbing the UF_6 and molybdenum fluoride (by raising the temperature to $>150^{\circ}\text{C}$ and passing fluorine through the bed), and passing desorbed gas through a fixed bed of MgF_2 , which is held at about 150°C . The MgF_2 sorbs the molybdenum but allows the purified UF_6 to pass through into the cold traps.

The NaF pellets used in the high-temperature zone must be replenished periodically since they accumulate fission products and corrosion products. This discard constitutes one of the waste streams. The two NaF sorption zones may be integrated into a single unit, one zone on the other, and, as NaF is discharged from the lower zone, it can be replenished by NaF from the low-temperature zone, which is in turn fed with fresh NaF pellets. Such a system has been used effectively in pilot plant operations, and a similar system would be desirable for the MSBR processing facility.^{17,18} In our concept, the system has a movable bed of NaF pellets, and, after each sorption cycle, some of the lower NaF is mechanically ejected to waste. Annular design with air cooling would probably be necessary to allow removal of fission product heat (Fig. 7).

As the UF_6 leaves the last sorption trap it must be collected and ultimately used as feed for the fuel-reconstitution step. This is done by collecting the UF_6 in cold traps. Two cold traps are connected in series. The first, or primary trap, is operated at about -40°C , and the second, or backup trap, is operated at about -60°C . The principal design consideration is the heat transfer rate. Conventional designs are available for such units in which there are internal cooling fins for collecting UF_6 , and Calrod heaters for vaporizing the UF_6 for removal (Fig. 8). Uranium hexafluoride from the heated cold trap is fed directly to the reduction process. This calls for at least three primary cold traps for continuous operation: one for collecting UF_6 , one for feeding UF_6 , and one in transition between these two functions.

A fixed bed of NaF at ambient temperature very effectively removes trace amounts of UF_6 from gas streams. Such beds are used as backup UF_6 traps in the fluorine exhaust from the cold traps and in other process streams that might contain UF_6 . Uranium is recovered from such traps by using the NaF as charge material for the main absorption beds.

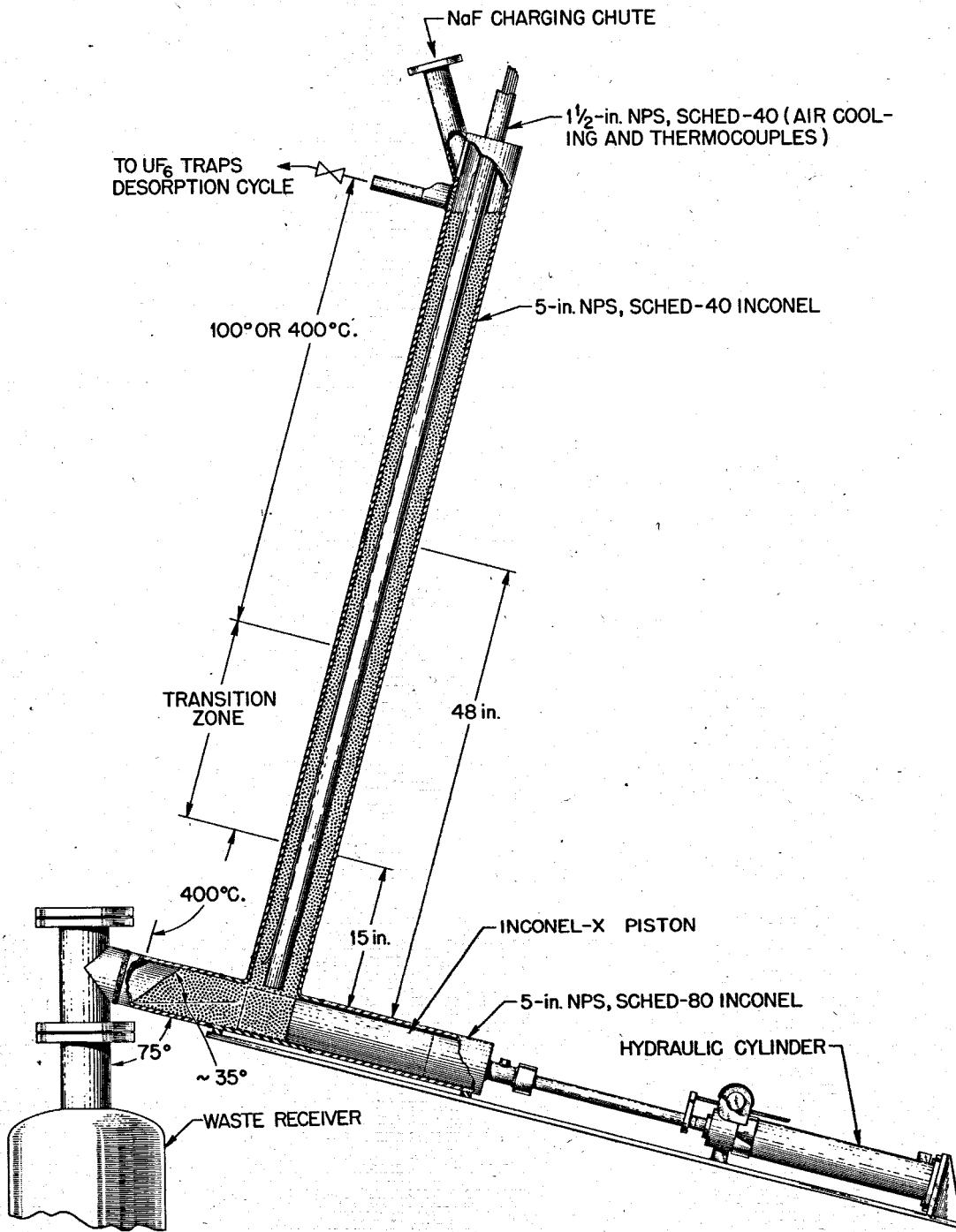


Fig. 7. Movable-Bed Temperature-Zoned Absorber. When the lower zone of the bed becomes loaded with fission products, the hydraulic cylinder operates the piston to discharge that portion of the bed into the waste carrier. Fresh NaF is added at the top. This apparatus has already been tested in the ORNL pilot plant.

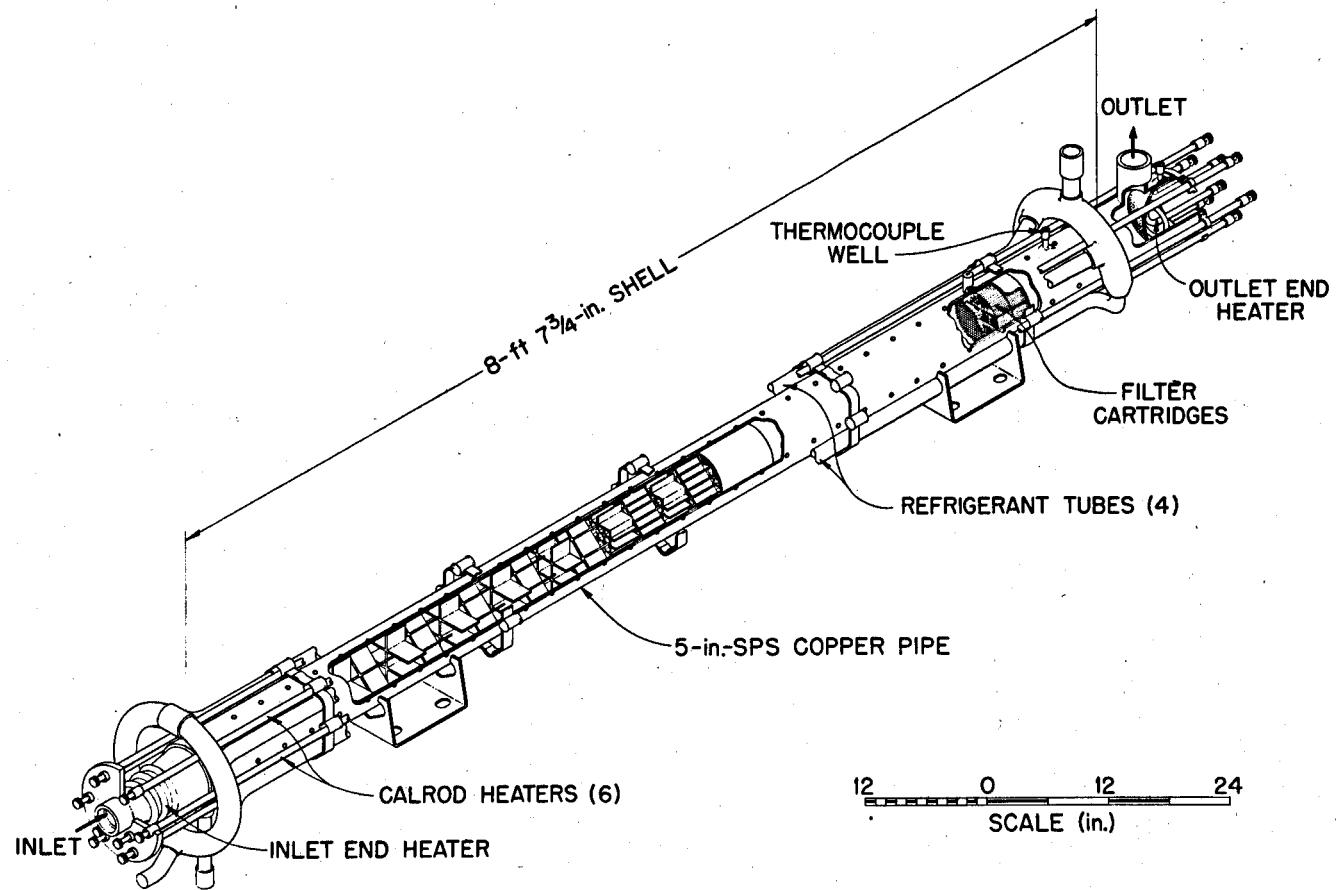


Fig. 8. Cold Trap for UF_6 Collection. This design has already been successfully used in the ORNL pilot plant.

Since there is excess fluorine in the fluorinator off-gas (33-l/3% utilization), fluorine is recycled. This recycle contains some fission products, so it is necessary to remove a side-stream (10%) to prevent their buildup. Fresh fluorine is used for the desorption step, for all processing in the blanket section, and for fluorine makeup.

Vacuum Distillation

The vacuum distillation step applies to the fuel stream only and is used to separate the carrier salt from the fission products after the uranium is removed by fluorination (see above). The LiF and BeF_2 volatilize, leaving fission products in the still bottoms. This residue consists largely of rare-earth fluorides. If the relative volatilities of the fission products, compared with the volatilities of LiF and BeF_2 , are low, then a good separation can be achieved in a single-step distillation without rectification.

At the average operating temperature (about 650°C) of the MSBR itself, the solubility of rare-earth fluorides in fuel salt is only a few mole percent; however, at 1000°C , the solubility in LiF alone is about 50 mole % for the more insoluble compounds, for example, LaF_3 , PrF_3 , and CeF_3 . Other rare-earth (RE) fluorides have even higher solubilities at this temperature.⁵ Physics calculations on the 58-day-cycle MSBR indicate that at equilibrium the molar ratio of LiF:(RE) F_3 in the fuel is about 1400, a number considerably greater than the 1:1 ratio permitted by the solubility limit at 1000°C . It is therefore apparent that, based on solubility data alone, distillation at about 1000°C can tolerate an extremely large rare-earth concentration factor before precipitation occurs. The design of the distillation unit is concerned primarily with determining the appropriate configuration that will permit a large fission-product concentration factor and, at the same time, provide adequate heat-removal capability for the short-cooled fuel.

The still design developed for this study is shown in Fig. 9. It is charged initially with 4 ft^3 of LiF that has the same isotopic composition as that in the reactor fuel; this volume fills the tubes to a depth of about an inch above their tops. The pressure above the LiF pool is reduced

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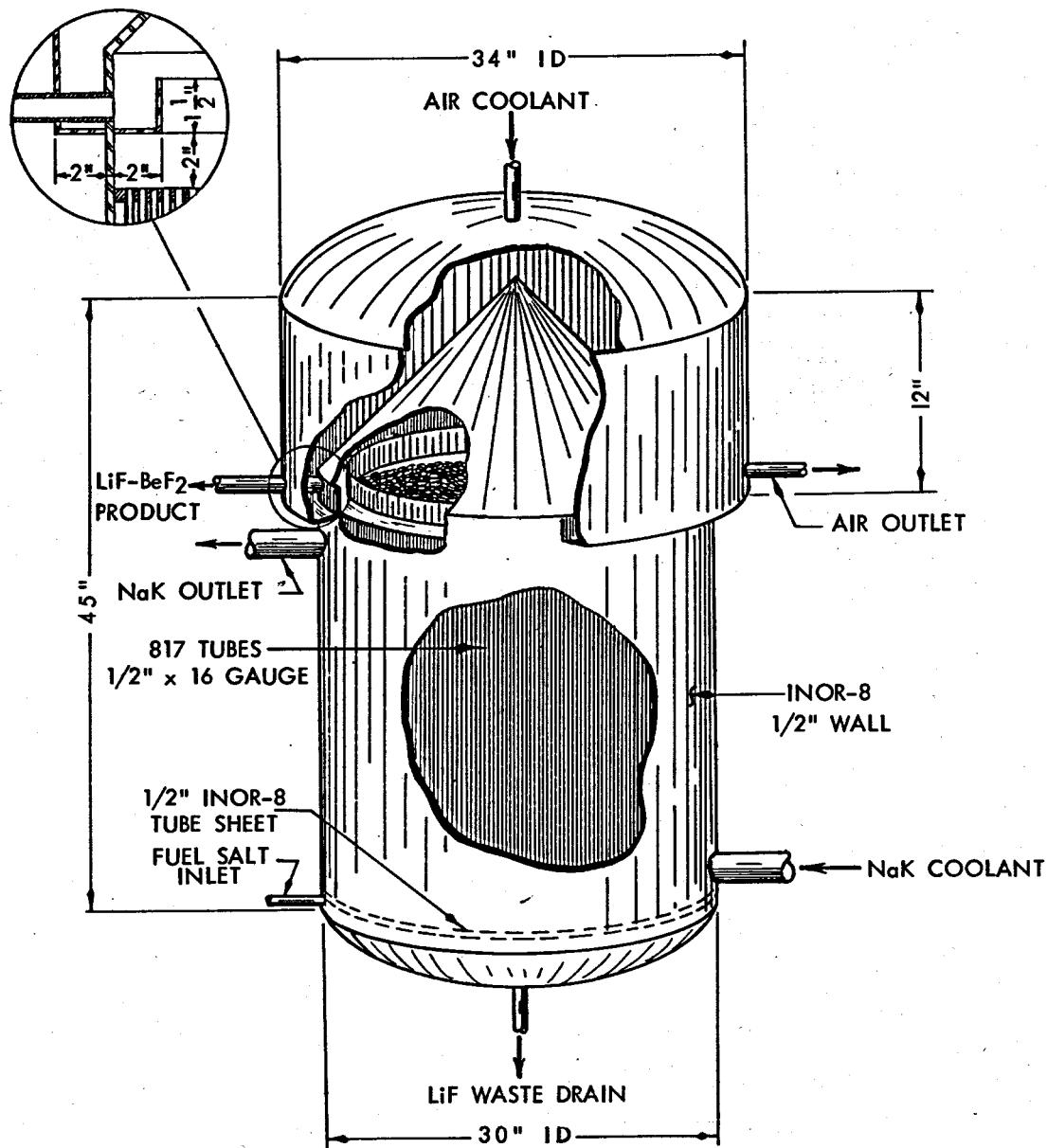


Fig. 9. Vacuum Still for MSBR Fuel. Barren fuel-carrier flows continuously into the still, which is held at about 1000°C and 1 mm Hg. LiF-B₂F₂ distillate is removed at the same rate that salt enters, thus keeping the volume constant. Most of the fission products accumulate in the still bottoms. The contents are drained to waste storage when the heat generation rate reaches a prescribed limit. This concept of the vacuum still has not been tested.

to 1 to 2 mm of Hg by evacuating the product receiver (see Fig. D-3 in Appendix D), and the temperature is adjusted and held at about 1000°C . Fluorinated fuel salt is continuously admitted to the LiF pool in the still, and distillation is allowed to proceed at the same rate as the inlet fuel rate so that there is no net volume change.

The operating principle is to allow the rare-earth fission products, which have a much lower vapor pressure than either LiF or BeF_2 , to concentrate in the still. The liquid pool in the still reaches an equilibrium concentration in LiF and BeF_2 , and the two components then distill at the rate at which each is entering the still. The condenser is a conical region located just above the evaporating surface; it is kept at about 850°C by forced convection of heated air. Distillate is collected in a circular trough and drained to a product receiver. The still is operated in this way until the heat-generation rate due to fission-product decay reaches the heat-removal capability of the NaK cooling system. This point occurs after about 67 days of continuous operation, at which time the fission product concentration in the still is about 14 mole %, a value considerably less than the approximately 50 mole % solubility limit at 1000°C . At this time, the contents of the still are drained to a permanent waste receiver, and the cycle is repeated. These calculations are conservative since they are based upon gross fission-product heat release and do not subtract the effect of those fission products removed or deposited in the reactor before chemical processing.

The attractive feature of carrying out the distillation in this way is that it minimizes the volume of expensive LiF relegated to waste. Since distillation is carried out after fluorination, less than 0.1% of the uranium removed from the reactor should enter the still, and at 1000°C the vapor pressure of UF_4 is favorable to the recovery of a significant portion of this fraction, reducing the overall uranium loss. Beryllium fluoride losses should be insignificant because the vapor pressure of BeF_2 at 1000°C is about 100 times that of LiF. Distillation probably will not effect decontamination from CsF and RbF; each of these compounds has a greater vapor pressure than either BeF_2 or LiF. Because rare earths are concentrating in the still as a function of time, their decontamination factor in the product will decrease with time. It is not possible with existing data to assess the magnitude of this effect.

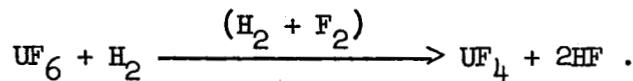
Detailed calculations for the still design are given in Appendices C and D. Vapor pressure data for principal components of the fuel stream are included in Appendix F.

Distillate from the still is collected in an evacuated tank operating at still pressure. When filled, the receiver is isolated from the still, and the LiF-BeF₂ mixture is transferred by gravity flow or pressurization to the reduction and fuel-makeup operations.

An in-cell waste receiver is provided for the initial cooling of the still residue before transfer to the underground waste-storage facility. The tank has a 4-ft³ volume, allowing a one-cycle delay (about 67 days) inside the cell where heat is conveniently removed by the circulating NaK coolant. During this delay, the heat generation rate decreases from 3.2×10^7 Btu/hr to 6.8×10^6 , imposing less stringent design requirements in the permanent waste receiver. The interim receiver is a shell-and-tube type, similar to the still; however, no condensing surface or provisions for air cooling are needed.

Reduction of Uranium Hexafluoride and Reconstitution of the Fuel

The combined UF₆ streams, that form the fuel salt and the fertile salt, are reduced to UF₄, and only a sufficient amount to maintain criticality is returned to the reactor. The excess ²³³U from the fertile stream is sold. The usual method for this reduction has been by reaction with the excess H₂ in an H₂-F₂ flame:



This reduction is carried out in a tall column into which UF₆ and H₂ are introduced into an H₂-F₂ flame, and dry UF₄ powder is collected as the product. However, according to our proposal, a more convenient method for preparing UF₄ for the MSBR is by reducing the UF₆ to UF₄ with H₂ after the UF₆ is dissolved in the molten salt. There are some experimental data indicating the feasibility of such a process; however, the kinetics of the absorption and of the reduction must be further investigated.²⁶ It is possible that this two-step process could be carried out continuously.

in a single column in which the molten salt flows upward, the UF_6 is introduced and dissolved in the bottom of the column, and the H_2 is introduced at an intermediate point to reduce the UF_6 (Fig. 10). Some of the reconstituted salt has to be recycled to the column to provide enough dissolved uranium for proper UF_6 absorption.

Off-Gas Processing

Most of the off-gas from the processing plant comes from the continuous fluorinators; smaller amounts are formed in various other processing vessels. The gases are processed to prevent the release of any contained fission products to the atmosphere. Excess fluorine used in the fluorinators is recycled through a surge chamber by a positive displacement pump, and a small side-stream of the recycling fluorine is sent through a caustic scrubber to prevent gross buildup of fission products. Each of the processing vessels and holdup tanks have off-gas lines which lead to the scrubber for removing HF, fluorine, and volatile fission products.

The scrubber operates as a continuous, countercurrent, packed bed with recirculating aqueous KOH. A small side-stream of KOH solution is sent to waste, and the scrubber off-gas is contacted with steam to hydrolyze fission products such as tellurium. A filter removes the hydrolyzed products. The noncondensable fission products are sent to the off-gas facility for gases generated by the reactor.

Waste Storage

Four waste streams requiring storage leave the processing facility: (1) NaF and MgF_2 sorbent from the UF_6 purification system, (2) aqueous waste from the KOH scrubber, (3) molten-salt residue from the distillation unit, and (4) molten salt from fertile stream discard. The aqueous waste stream is small, and it is assumed that adequate capacity exists in the system for storing reactor waste. The other three wastes are stored in separate underground facilities adjacent to the processing cells. Since the values in the waste from the fertile stream - ^{7}Li , thorium,

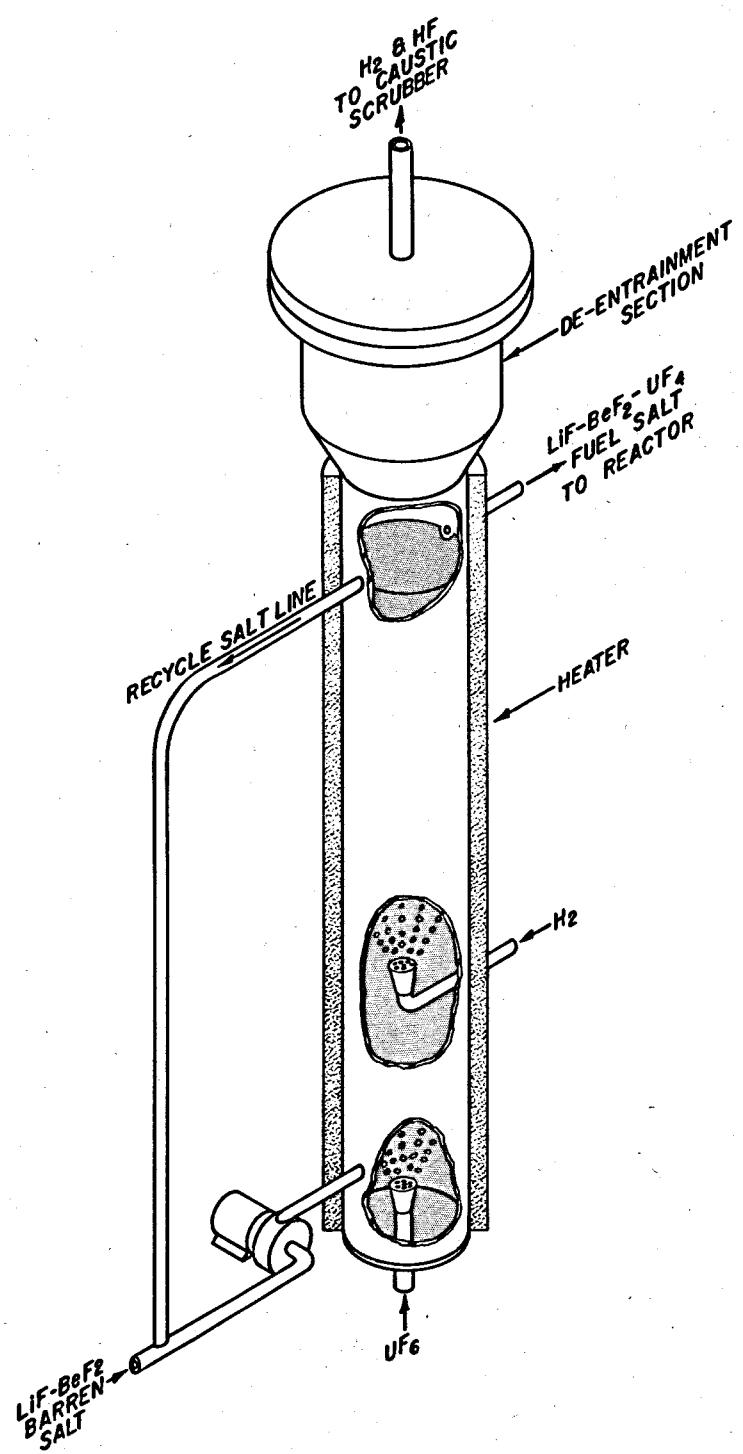


Fig. 10. Continuous Reduction Column. Barren salt and UF_6 enter the bottom of the column, which contains circulating LiF - BeF_2 - UF_4 . The UF_6 dissolves in the salt, aided by the presence of UF_4 , and moves up the column where it is reduced by hydrogen. Reconstituted fuel is taken off the top of the column and sent to the reactor core. Preliminary data indicate that this design is promising.

and ^{233}U - will be worth recovering at some future time, some very tentative ideas about how they may be recovered are presented at the end of this section.

NaF and MgF_2 Pellets

Spent NaF and MgF_2 pellets, which retain the volatile fission and corrosion product fluorides from the UF_6 gas stream, are stored in 10-in.-diam by 8-ft-high stainless steel cylinders in a concrete vault adjacent to the still-residue waste tank. (See Dwg. No. 58080D in Appendix F.) The cylinders are loaded inside the processing cell and transferred to the underground area at approximately 90-day intervals. The vault is designed to contain a 5-year collection of cylinders (160 cylinders at 100% plant factor); after 5 years, the older cylinders are removed and transferred to a permanent underground storage site. The integrated heat generation when the vault contains 160 cylinders is about 1.73×10^6 Btu/hr. Forced circulation of about 11,300 scfm of air at a temperature rise of 125°C is used to remove this energy. The containers are constructed with a hollow core, allowing coolant to pass through the cylinder as well as over the outside.

Aqueous Waste from Off-Gas Scrubber

This waste, fission products in a strong solution of KOH, will be stored along with other aqueous wastes from the reactor system and represents an insignificant contribution to the total amount of aqueous waste.

Fuel-Stream Waste

Residue from the vacuum still is stored in bulk in a facility similar to one evaluated previously by Carter and Ruch.¹⁴ A single, large tank equipped with adequate cooling tubes and adequate for a 30-year accumulation of waste is provided. The 30-year capacity was chosen since that is the expected lifetime of the reactor. After filling, the salt might remain in the tank for additional decay or be disposed of by whatever method is currently acceptable.

Decay heat is removed by forced air convection. The heat load (Btu/hr) continually increases over the filling period but decreases rather sharply when no further additions are made (Fig. E-1, Appendix E). The time behavior of the integrated specific heat generation rate ($\text{Btu hr}^{-1} \text{ ft}^{-3}$) for a 5-year collection period is shown in Fig. 11. This is a smoothed curve for 4-ft³ additions every 67.4 days, followed by an extended decay period during which no waste is added to the tank. The upper portion of the curve might be extrapolated with little error to accommodate longer filling periods. Figure 11 shows the specific heat generation for the case of no dilution with inert salt; however, during the initial stages of filling, it is necessary to add an inert diluent, for example, NaF-KF eutectic mixture, to lower the specific rate to a tolerable value. It was calculated that 264 ft³ of diluent is required for the 520 ft³ of LiF-fission product residue to be collected over the 30-year period. The fission products being collected exhibit the decay behavior shown in Fig. 2, which is representative of gross fission products and does not account for those that have been removed by processing or other mechanisms.

Fertile-Stream Waste

The fertile-stream discard is also stored in a large underground tank, adequate for 30 years of waste collection. The tank is 13.5 ft in diameter and 13.5 ft high. Since uranium is removed from the blanket on a rather fast cycle, the fission rate in the blanket is low, making the waste activity several orders of magnitude less than that of the fuel waste. Cooling is provided by natural air convection around the tank and through cooling tubes. All metal surfaces are expected to be coated with a layer of frozen salt that will furnish excellent corrosion protection.

The integrated heat production rate due to fission products for the 30-year period is 5.91×10^4 Btu/hr. Since this energy is associated with 1783 ft³ of LiF-ThF₄ mixture, the moderate specific rate of $33.1 \text{ Btu hr}^{-1} \text{ ft}^{-3}$ presents no design problems. When first removed from the reactor, the heat production rate of the waste is about $1600 \text{ Btu hr}^{-1} \text{ ft}^{-3}$, but this value decreases by a factor of 10 in about 4 days.

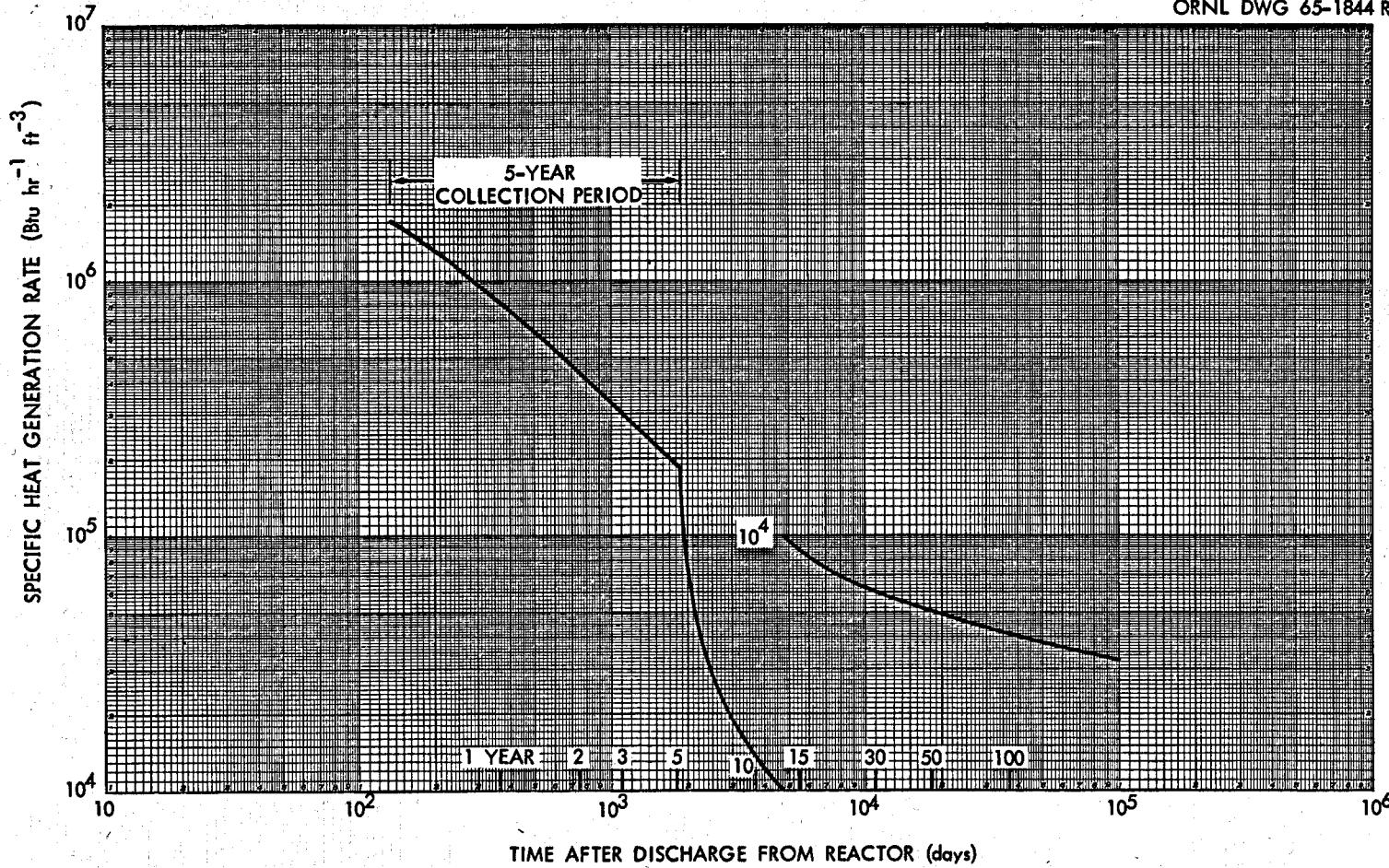


Fig. 11. Specific Heat Generation Rate of Fuel-Stream Fission Products in Waste Tank. Waste is accumulated in 4 ft³ batches every 67.4 days. It is then held in the processing cell another 67.4 days for further cooling before draining to the waste tank. The reactor operates on a 58-day cycle at 2160 Mw (thermal).

Cooling System for the Waste

As mentioned above, the waste-storage system is designed for cooling by forced air draft. Air is easy to handle, compatible with construction materials for extremely long times, and presents a minimum hazard in case of contamination by the waste. Blowers, capable of supplying 76,000 scfm at a pressure drop of about 30 in. water, are located upstream from the waste vaults. The air is forced through the vaults and cooling tubes in the waste tank and is exhausted to the atmosphere via a tall stack, which also disposes of gases exhausted from the reactor system. The exit duct contains the necessary radiation-monitoring instruments and absolute filters for removing particles.

Possible Ultimate Treatment of Waste from Fertile Stream

Since a 30-year accumulation of waste from the fertile stream will contain recoverable values (^7Li , ^{233}U , and thorium), not too highly contaminated with fission products, it may be worthwhile to consider a recovery system before relegating this waste to permanent burial. Any significant uranium value would probably be recovered by fluorination, but the recovery of thorium and lithium requires further process development. A potential method for Li-Th separation is the incompletely investigated HF-dissolution process,¹⁵ based on the principle of leaching LiF from ThF_4 and rare-earth fission products with anhydrous hydrofluoric acid. This process, however, leaves the thorium contaminated with fission products, making it necessary to resort to an aqueous system (solvent extraction by the Thorex process) or to develop a thorium recovery process that utilizes fluoride chemistry. From a purely economic viewpoint, thorium would be retained in the waste tank until its recovery became more economical than mining new thorium.

On a 30-year discard cycle, an entire fertile-stream inventory of ThF_4 and LiF will accumulate in the waste tank. In addition, the waste will contain ^{233}U , ^{233}Pa , and fission products in amounts that depend on the breeding ratio, efficiency of the fluorination step, and the blanket power. The uranium loss is based upon a 90% efficiency in fluorination, a value believed to be conservative. The largest loss of fissionable

material, however, is through protactinium, which we assume to be non-volatile as the fluoride and which consequently is discarded in direct ratio to its concentration. The amount of fission products is calculated for a blanket power of 62 Mw (thermal) and an 80% plant factor; to account for those fission products that have volatile fluorides or which are plated out on parts of the system, a nominal figure of 80% is used for the fraction that finally reaches the waste tank. The 30-year inventory of the waste tank is shown in Table 2.

Table 2. Inventory in Fertile-Stream Waste Tank After 30 Years of Collection

Waste volume = 1783 ft³

	Amount (kg)	Unit Value (\$/kg)	Value (10 ⁶ \$)
Th (as ThF ₄)	141,200	10	1.41
⁷ Li (as LiF)	10,400	120	1.25
²³³ U + ²³³ Pa (as UF ₄ + PaF ₄)	116	12,000	1.39
Fission products (as fluorides)	450		
			4.05

In view of the figures in Table 2, the design presented here for the fertile-stream waste system is not optimal. Nearly 99.5% of the ²³³U + ²³³Pa value in the waste tank is attributable to ²³³Pa discard; this loss can be reduced to negligible proportions by providing in-cell decay storage followed by refluorination to recover the daughter ²³³U. For example, a 1-year holdup (about 60 ft³) would allow more than 99.995% of the ²³³Pa to decay to ²³³U. It probably would not be necessary to provide additional fluorination equipment for this volume because the refluorination could be scheduled in existing equipment during reactor shutdown.

Flow Control of the Salt Streams

Flow control of the molten salt streams is by freeze valves coupled with a controlled pressure drop. This can be achieved by the simple freeze valve currently used in the MSRE (Molten Salt Reactor Experiment),

coupled with a flow restriction such as an orifice or venturi. A dynamic freeze valve in which a controlled layer of salt is built up in a cooled section of the line is being investigated; if successful, it will allow a greater ease of flow control.²⁷

Removal of Decay Heat

Heat removal is a major problem in all process vessels that contain short-cooled, highly irradiated fuel. In many cases, the heat flux and operating temperature will be high, making it difficult to use water or air as primary coolants. Water has an additional disadvantage in that it is incompatible with the process fluids, creating a hazard should there be a leak in the salt lines. Therefore, a sodium-potassium eutectic mixture, 22.3-77.7 wt % Na-K, was chosen as the primary coolant for process vessels at temperatures above 500°C where large amounts of decay heat must be removed. This coolant also has the capability of initially heating the system to 600°C for startup purposes. Air was chosen as the coolant for low heat fluxes at temperatures less than 500°C.

Sampling the Salt and Off-Gas Streams

A rather complicated mechanism is required to remove analytical samples from a highly radioactive molten-salt system. A technique similar to that tested for the MSRE will be used,²⁸ and the off-gas streams will be sampled conventionally.

Shielding, Maintenance, and Repair of Equipment

All process equipment that handles material that approaches the radiation level in the reactor core will be shielded by about the same amount of shielding as for the reactor, and maintenance will be indirect. Process vessels in this area needing repair will be removed and sent to a decontamination facility before repair. These include the fuel-stream delay tanks, fluorinator, sorption-desorption systems, and distillation system. All other equipment for processing radioactive materials is

contained in an area of direct maintenance with less shielding. In the direct-maintenance area, decontamination for maintenance is achieved by a molten-salt flush and an aqueous wash.

Materials of Construction

All process vessels and lines in contact with molten salt are made of INOR-8 except the fluorinators, which are made of a special material, Alloy 79-4.²⁸ Vessels and lines that contain fluorine or fluoride-bearing gases are made of Inconel or Monel, and cold traps are made of copper. Other equipment is made of appropriately compatible material.

General Operating Policy

The overall policy for operating the fluorinator, vacuum still, NaF beds, and related equipment is based on the projected simplicity of operation and small size of the equipment. The system is designed for a campaign-type operation of one month's duration, without shutdown except for emergency maintenance. There will be no access to the operating areas during this period. At the end of the operating period, the entire system will be closed down, routine maintenance accomplished, feed hoppers replenished, accumulated waste transferred to waste storage, etc. The operating cycle is then repeated.

PROCESS DESIGN

Process equipment for this design study was patterned as much as possible after previously designed and tested equipment. Each major component was studied for its application to the continuous processing requirements of the MSBR to ensure a practical design. Detailed designs were made to the extent that overall size, configuration, heat transfer requirements, flow rates, etc., were defined to allow a reasonable estimate of equipment cost. The waste system was considered separately, and liberal use was made of a previous, similar study¹⁴ for its design and cost data.

Fuel Stream

Design Criteria

Design criteria for process equipment were based on past experience. Equipment capacity was based on handling a flow rate 30% larger than that required for 100% continuous operation as specified by reactor physics calculations. Pertinent basic data adopted for this design are as follows:

Salt flow rate: 15 ft³/day

Pressure: 2 atm

Density of pelleted sorbents, NaF and MgF₂: 1 g/cc

Temperature range for NaK coolant:

As coolant: 200°C

As heat source: 800°C

Coolant temperature rise in heat transfer operations: 300°C

Normal convective heat transfer coefficient: 10 Btu hr⁻¹ ft⁻² °F⁻¹

Electrical heat needed: 1.5 kw per ft² of longitudinal area for temperatures higher than 500°C

Fission product heat: 50% of total is removed with volatile fluorides in the fluorination step

Major Process Equipment

There are 41 major pieces of fuel-stream process equipment (Fig. 12). Most of the equipment design is straightforward and based on conventional techniques.²⁹⁻³² Each component shown in the processing flowsheet (Fig. 12) is listed below, with its purpose, design basis, and description. The identifying number accompanying the equipment name corresponds to the circled numbers in the figure. Design calculations that involve unusual techniques or complexity are shown in the appendices.

1. Flow Control

Purpose: The flow-control device meters the flow of a molten salt stream to or from a process vessel.

Description: A piece of process pipe, 1/2-in., sched-40, 1 ft long. Pipe is jacketed with 1-1/2-in. sched-40 pipe made of INOR-8 and has two coolant connections.

2. Coolant Tank

Purpose: To allow delay for fission-product decay of the molten-salt stream from the reactor core before its introduction to the fluorinator.

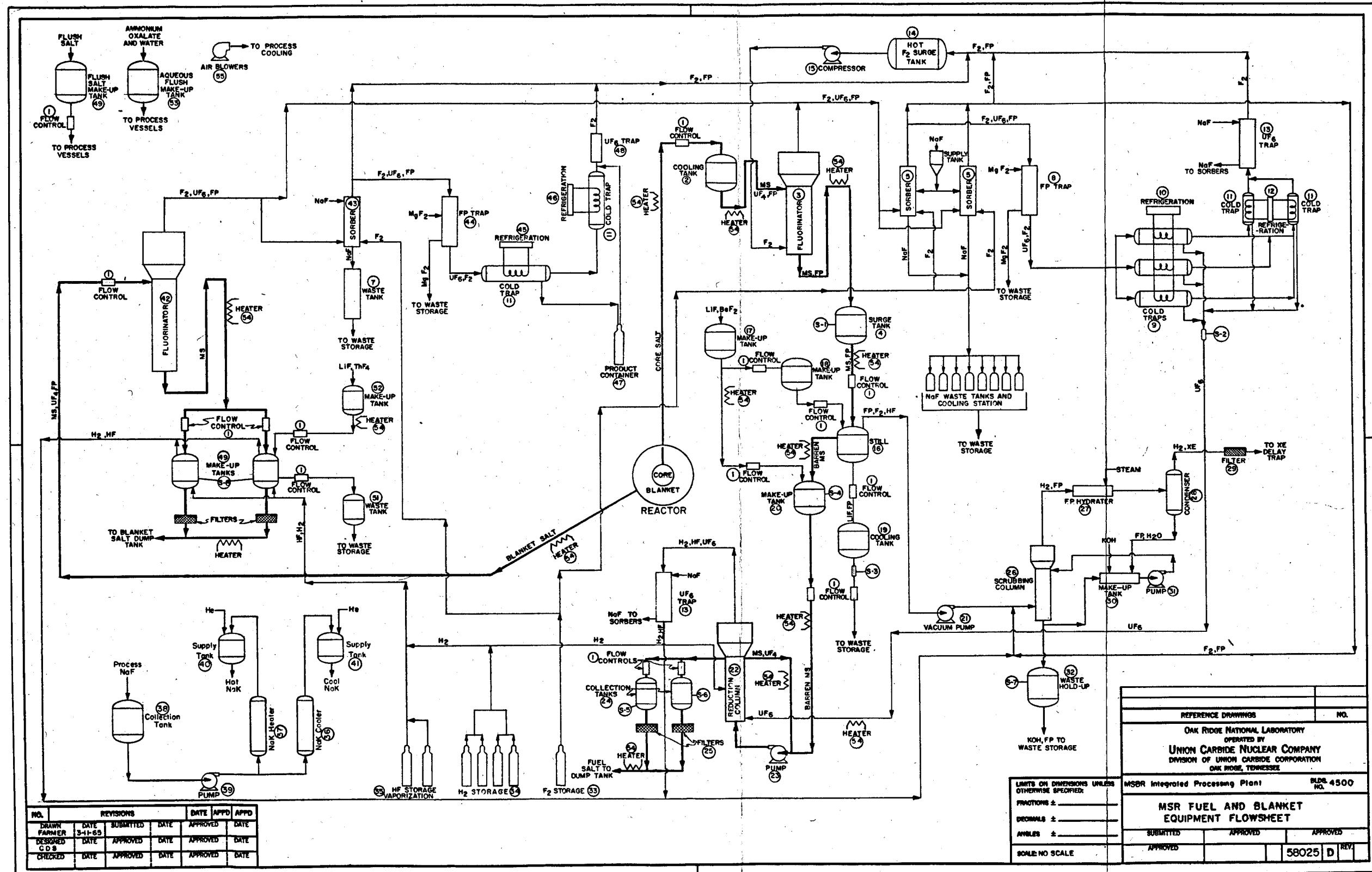


Fig. 12. Major Components in Fuel- and Fertile Stream Process.

Design Basis:

Type of tank: Heat exchanger with well-mixed contents

Salt residence time: (See Appendix A)

Temperature: 600°C

Coolant: NaK

Heat load: $5.14 \times 10^4 \text{ Btu hr}^{-1} \text{ ft}^{-3}$

Material of construction: INOR-8

Description: A tank having 22.5 ft³ of liquid volume, with bayonet coolers and two pipe connections.

3. Fuel-Salt Fluorinator

Purpose: To remove all uranium and volatile fission products from the fuel salt by continuous fluorination.

Design Basis:

Salt residence time: 2 hr

Fluorine utilization: 33-1/3%

Maximum mass flow rate of F₂: 0.277 slpm/in.²

Temperature: 600°C

Coolant: NaK

Heat load: $5.31 \times 10^4 \text{ Btu hr}^{-1} \text{ ft}^{-3}$

Material of construction: Alloy 79-4

Description: A column made of 6-in. sched-40 pipe, 11-1/2 ft long, which is jacketed with an 8-in. sched-40 pipe. Column is expanded above jacketed section to 8-in. sched-40 pipe, 1 ft long. Column is supplied with 5 kw of electrical heat and has 5 pipe connections.

4. Surge Tank

Purpose: To allow molten-salt surge capacity between the fluorination and distillation steps.

Design Basis:

Temperature: 600°C

Surge capacity: 1 day's continuous flow (15 ft³)

Coolant: NaK

Heat load: $4.4 \times 10^4 \text{ Btu hr}^{-1} \text{ ft}^{-3}$

Material of construction: INOR-8

Description: A tank having 15 ft³ of liquid volume, with bayonet coolers and two pipe connections.

5. Movable-Bed Sorber

Purpose: To separate some of the volatile fission products and corrosion products from UF₆.

Design Basis:

Number of sorption zones: 3

Cooling load in high-temperature trap: $5.52 \times 10^5 \text{ Btu/hr}$

UF₆ load based on 12-hr cycle: 13.5 kg

NaF loading: 0.5 g UF₆/g NaF

Coolant: Air

Average temperature: 400°C in bottom zone, 100 to 150°C in two top zones

NaF Usage: 20% of one zone volume per day

Material of construction: Inconel

Description: An annular column made of two concentric pipes. The outer pipe is 10-in. sched-40, 8 ft long; the inner pipe is 6-in. sched-40, 8 ft long with the bottom 30 in. finned. Bottom mechanical solids ejection; 34 kw of electrical heat.

6. NaF Supply Tank

Purpose: To maintain a supply of NaF pellets for the movable-bed absorbers.

Design Basis:

NaF supply period: 30 days

NaF usage: 10.75 kg/day

Temperature: Ambient

Material of construction: Stainless steel

Description: A tank, 1-1/2 ft in diameter and 3 ft high, with a conical bottom and four 2-in. star valves.

7. NaF Waste Tanks and Cooling Station

Purpose: To provide short-term waste storage for the solid sorbents which contain sorbed fission products.

Design Basis:

Tank capacity: 107.5 kg NaF

Coolant: Air

Temperature: <400°C on surface

Capacity of cooling station: 8 tanks

Heat rate of cooling station: 2.0×10^6 Btu/hr (Appendix B)

Material of construction: Inconel

Description: Vessel made of two concentric pipes; outer pipe is 10-in. sched-20 pipe, 8 ft long; the inner pipe is 3-in. sched-20. Vessels supported in a metal frame.

8. Fission Product Trap

Purpose: To remove the fission product Mo from a UF₆ fission product stream by sorption on MgF₂.

Design Basis:

Temperature: 150°C

Coolant: Air

Capacity: 20 kg of MgF₂ pellets

Material of construction: Inconel

Description: Vessel made of two concentric pipes; outer pipe is 6-in. sched-40 by 5-1/2 ft long; inner pipe is 3-in. sched-40. Vessel heated by 5 kw of electrical heat.

9. Primary Cold Trap and UF₆ Vaporizer

Purpose: To serve as the primary cold trap for collecting UF₆ from the sorber effluent and to be the UF₆ feed station for the reduction column.

Design Basis:

UF₆ capacity: 13.5 kg
Temperature: -40°C to 100°C

Material of construction: Copper

Description: An internally finned vessel, 5 in. in diameter and 14 ft long, with 15 kw of electrical heat

10. Refrigeration Unit

Purpose: To provide coolant for the fuel-salt primary cold traps.

Description: Freon-type, rated at 60,000 Btu/hr at -45°C.

11. Secondary Cold Trap

Purpose: To act as a backup to the primary cold trap.

Description: An internally finned copper vessel, 6 in. in diameter and 6 ft long, with 5 kw of electrical heat.

12. Refrigeration Unit

Purpose: To provide coolant for the fuel-salt secondary cold traps.

Description: Freon-type, rated at 8,000 Btu/hr at -75°C.

13. UF₆ Trap

Purpose: To remove trace amounts of UF₆ that escape the secondary cold trap.

Design Basis:

Temperature: Ambient
NaF Capacity: 20 kg
Material of construction: Inconel

Description: A vessel made of 4-in. sched-40 pipe, 4 ft long.

14. Fluorine Surge Tank

Purpose: To provide a surge capacity for the recycle F₂ stream.

Design Basis:

Capacity: 1-hr supply
Temperature: Ambient
Material of construction: Inconel

Description: A tank, 1-1/2 ft in diameter and 4 ft long.

15. Fluorine Compressor

Purpose: To recycle fluorine from the fluorinator off-gas back to the fluorinator inlet.

Design Basis:

Type: Diaphragm with remote head
Capacity: 6 to 10 ft³/hr
Temperature: Ambient
Material of construction: Monel

Description: A diaphragm pump with remote head.

16. Vacuum Still

Purpose: To separate fuel-carrier salt from less volatile fission products.

Design Basis:

Pressure: 1 mm Hg
Temperature: 1000°C
Liquid volume: 4 ft³
Heat transfer: Appendix D
Distillation rate: 15 ft³ of salt per day
Coolant: NaK

Operation: Continuous addition of feed and continuous removal of distillate; periodic removal of residue.

Description: An INOR-8 vessel of shell-and-tube design, 2.5 ft in diameter and 2.5 ft high; 267 ft² of heat transfer area in liquid section and 6.9 ft² in condensing section; 45 kw of electrical heat.

17. Salt Makeup Tank

Purpose: To prepare barren salt in the nonshielded operating area for use as salt makeup.

Design Basis:

Temperature: 900°C
Capacity: 1 ft³
Material of construction: INOR-8

Description: A tank, 1 ft in diameter and 1.5 ft high, with 8 kw of electrical heat.

18. LiF Makeup Tank

Purpose: LiF makeup for the distillation unit.

Design Basis:

Temperature: 900°C
Capacity: 4 ft³
Material of construction: INOR-8

Description: A tank, 1.5 ft in diameter and 3 ft high, with 20 kw of electrical heat.

19. Coolant Tank for Fuel-Stream Waste

Purpose: To provide short-term storage and cooling of the waste stream from the vacuum still.

Design Basis:

Temperature: 1000°C

Capacity: 4 ft³

Heat removal: See Appendix D

Material of construction: INOR-8

Description: A tank having 4 ft³ of liquid volume, with bayonet coolers and 40 kw of electrical heat.

20. LiF-BeF₂ Makeup Tank

Purpose: To provide surge capacity and barren salt makeup for the molten salt fed to the reduction column.

Design Basis:

Temperature: 600°C

Capacity: 8 hr of retention

Material of construction: INOR-8

Description: A tank, 1.5 ft in diameter and 3.5 ft high, with 40 kw of electrical heat.

21. Vacuum Pump

Purpose: To provide vacuum for the vacuum still.

Design Basis:

Displacement: 40 cfm

Pressure: <50 μ Hg.

Material of construction: Steel

Description: Commercial, oil-pumped vacuum unit.

22. Reduction Column

Purpose: To reduce the purified UF₆ in the still distillate to UF₄. This reconstitutes the fuel.

Design Basis:

Temperature: 600°C

Residence time for dissolving: 30 min

Residence time for reduction: 1 hr

UF₆ rate: 26.9 kg/day

UF₆ mass flow rate: 7 g in.⁻² hr⁻¹

H₂ utilization: 50%

Material of construction: INOR-8

Description: A column, made of 12-in. sched-40 pipe, 8 ft high, expanded at top to 18-in. sched-40 pipe by 18 in. long. Heated by 50 kw of electrical heat.

23. Molten-Salt Pump

Purpose: To recycle molten salt in the reduction system.

Design Basis:

Type: Open bowl with helium purge
Temperature: 600°C
Capacity: 2 ft³/hr at 20-ft head
Material of construction: INOR-8

Description: Similar to pumps developed for Molten Salt Reactor Experiment.

24. Collection Tank for Recycle Fuel

Purpose: To provide surge capacity and a means of transferring molten salt to the reactor dump-tank.

Design Basis:

Temperature: 600°C
Capacity: 12-hr retention
Material of construction: INOR-8

Description: A tank, 2 ft in diameter and 2.9 ft high, with 27 kw of electrical heat.

25. Filters

Purpose: To remove particles from processed molten salt before its return to the reactor.

Design Basis:

Filtration area: 0.1 ft²
Material of construction: Porous nickel

Description: Porous metal filter in a 2.5-in.-diam by 1-ft canister, with 0.1 ft² area and a 2-kw electrical heater.

26. Off-Gas Scrubbing Column

Purpose: To remove volatile fluorides from waste-gas streams.

Design Basis:

Temperature: Ambient
Type: Countercurrent packed column
Packing: 1/2-in. Raschig rings
Source of fluoride gases: 10% of F₂ (derived from fuel-salt fluorinator), all the HF from reduction column, all the HF from blanket-salt sparging. The HF rates from these two sources are equal.

Gas mass flow rate: Less than 50% of flooding

Aqueous purge: 1 gal of aqueous KOH per hour

Cooling: Water

Material of construction: Monel

Description: A column, which is 3-in. sched-40 pipe, by 6 ft high, having an expanded section of 6-in. sched-40 pipe by 1 ft high on top. Column is jacketed with 5-in. sched-40 pipe, 6 ft high, and packed with 1/2-in. Raschig rings.

27. Fission Product Hydrator

Purpose: To hydrate all volatile fission-product fluorides that leave the scrubbing column.

Design Basis:

Temperature: 100°C

Steam rate: Equivalent to 1 gal of H₂O per hr

Material of construction: Monel

Description: A vessel which is 4-in. sched-40 pipe by 2 ft high with a 1-kw electrical heater.

28. Condenser

Purpose: Remove the condensable material from gases leaving the fission-product hydrator.

Design Basis:

Temperature: Ambient

Coolant: Air

Material of construction: Monel

Description: A 4-ft² heat exchanger.

29. Absolute Filter

Purpose: To remove particles from the noncondensable gases leaving the condenser.

Design Basis:

Temperature: Ambient

Filter area: 0.1 ft²

Filtering medium: Fiberglas

Material of construction: Stainless steel

Description: A commercial unit.

30. Makeup Tank for KOH

Purpose: To provide surge capacity and makeup volume for the scrubbing column.

Design Basis:

Temperature: Ambient

Capacity: 8 hr of scrubber flow

Material of construction: Monel

Description: A tank, 1.5 ft in diameter and 2 ft high.

31. Pump for Solution in Off-Gas Scrubber

Purpose: To circulate the aqueous KOH in the scrubbing system.

Design Basis:

Temperature: Ambient

Type: Canned rotor

Capacity: 7 gal/hr at 5-ft head
Material of construction: Monel

Description: Commercial unit.

32. Aqueous-Waste Tank

Purpose: To provide short-term holdup of aqueous waste.

Design Basis:

Temperature: Ambient
Capacity: 10 days
Material of construction: Monel

Description: A tank, 3 ft in diameter and 5 ft high.

33. Fluorine Storage and Supply

Purpose: To provide the primary fluorine supply for the system.

Type: Tank and trailer.

Description: Tank trailer containing gaseous fluorine.

34. Hydrogen Storage and Supply

Purpose: To provide the primary H₂ supply for the system.

Design Basis:

Type: 200-scf cylinders with pressure-reducing station
Capacity: 48-hr supply
Material of construction: Steel

Description: Four high-pressure cylinders; commercial units.

35. Hydrogen Fluoride Storage and Supply

Purpose: To provide HF supply for the system.

Design Basis: Continuous supply from 200-lb cylinders in constant-temperature steam bath.

Description: Two commercial cylinders.

36. NaK Cooler

Purpose: To remove heat from the NaK coolant.

Design Basis:

Type: Outside, air-cooled
NaK capacity: 5251 ft³/hr
NaK temperature: Enter 500°C, exit 200°C
Average air temperature: 27°C
Material of construction: Steel

Description: Air-cooled heat exchanger with 9200 ft² of heat transfer area and a 60-hp fan.

37. NaK Heater

Purpose: To heat the NaK so it can be used as a heating fluid for startup and checkout.

Design Basis:

Type: Heat exchanger, using electric heat

NaK capacity: 180 ft³/hr

NaK temperature: Enter at 500°C and leave at 800°C

Temperature on hot side of exchanger: 900°C

Material of construction: Stainless steel

Description: A heat exchanger with 270 ft² of heat transfer area and 355 kw of electrical heat.

38. NaK Collection Tank

Purpose: To collect NaK coolant and to serve as surge capacity.

Design Basis:

Temperature: 500°C

Volume: 200 ft³

Material of construction: Steel

Description: A tank, 5 ft in diameter and 10.5 ft long.

39. NaK Pump

Purpose: To circulate the NaK through process equipment and heat exchangers.

Design Basis:

Type: Centrifugal

Capacity: 650 gal/min with 50-ft head

Temperature: 500°C

Material of construction: Stainless steel

Description: A commercial centrifugal pump, powered by a 15-hp motor.

40. NaK Supply Tank

Purpose: To provide surge capacity for NaK when it is used as a heating fluid.

Design Basis:

Temperature: 800°C

Volume: 10 ft³

Material of construction: Stainless steel

Description: A tank, 2 ft in diameter and 4 ft long, with 36 kw of electrical heat.

41. NaK Supply Tank

Purpose: To supply NaK coolant to process equipment

Design Basis:

Temperature: 200°C

Volume: 100 ft³

Material of construction: Steel

Description: A tank, 3.5 ft in diameter and 10 ft long.

Fertile Stream

Design Criteria

Process equipment for the fertile stream was also designed for 30% greater capacity than that specified by reactor physics calculations for 100% continuous operation. The design rate is 105 ft³ of fertile salt per day. Other basic design values are the same as listed above in the design criteria for the fuel stream.

There are 14 major items of process equipment; the function of each is shown in Fig. 12. The listing below gives a description of each item, and the identifying numbers correspond to the circled numbers of Fig. 12.

42. Blanket-Stream Fluorinator

Purpose: To remove uranium from the blanket salt by continuous fluorination.

Design Basis:

Salt residence time: 2 hr

Fluorine utilization: 33-1/3%

UF₆ rate: 15.15 g-moles/day

Temperature: 600°C

Coolant: NAK

Maximum mass flow rate of F₂: 0.277 slpm/in.²

Coolant load: 42 kw

Resistance-heating load: 42 kw

Material of construction: Alloy-79-4

Description: A column, 12-in. sched-40 pipe, 16.5 ft high, which is jacketed with 14-in. sched-40 pipe, 16.5 in. high. Column is enlarged at top to 24-in. sched-40 pipe, 18 in. high. Heated by 42 kw of low-voltage alternating current applied to 14 ft² of nickel electrodes; supplemental heat supplied by 4 kw units.

43. Movable-Bed Sorber

Purpose: To separate some of the volatile fission products and corrosion products from UF₆.

Design Basis:

Number of sorption zones: 3

UF₆ load, based on 24 hr of operation: 221 g

NaF loading: 0.2 g UF₆ per gram of NaF

Coolant: Air

Average temperature: 400°C in bottom zone, 100 to 150°C in two top zones

NaF usage: 20% of one zone volume per day

Material of construction: Inconel

Description: Column made of two concentric pipes, outer one is 5-in. sched-40 pipe, 3 ft high; inner one is 1.5-in. sched-40, 3 ft high.
Mechanical ejector for spent NaF; 5 kw of electrical heat.

44. Fission-Product Trap

Purpose: To remove fission products from a UF₆-fission-product stream by sorption on MgF₂.

Design Basis:

Temperature: 150°C

Capacity: 5 kg of MgF₂ pellets

Material of construction: Inconel

Description: A vessel which is 4-in. sched-40 pipe, 2 ft high, warmed by 2 kw of electrical heat.

45. Refrigeration Unit

Purpose: To provide cooling for the blanket-stream primary cold trap.

Description: Freon-type with 24,000 Btu/hr cooling capacity at -45°C.

46. Refrigeration Unit

Purpose: To provide cooling for blanket-stream secondary cold trap.

Description: Freon-type with 4,000 Btu/hr cooling capacity at -75°C.

47. Product Receiver

Purpose: To receive the UF₆ product from the blanket-stream cold traps.

Design Basis:

Temperature: Ambient

Capacity: 10 days' production

Material of construction: Monel

Description: A vessel, 4-in. sched-40 pipe, 2 ft long.

48. UF₆ Trap

Purpose: To remove trace amounts of UF₆ from the cold-trap off-gas.

Design Basis:

NaF capacity: 5 kg

Temperature: Ambient

Material of construction: Inconel

Description: A vessel, 4-in. sched-40 pipe, 2 ft long

49. Makeup Tank

Purpose: To act as surge tank, molten-salt makeup tank, flush-salt makeup tank, and means of molten-salt transfer for the purified blanket salt.

Design Basis:

Temperature: 600°C

Capacity: 12 hr retention

Material of construction: INOR-8

Description: A tank, 3.5 ft in diameter and 6.5 ft high, with 105 kw of electrical heat.

50. Filter

Purpose: To remove particles from the processed molten salt prior to its return to the reactor.

Design Basis:

Filtration area: 1 ft²

Material of construction: Porous nickel

Description: A porous metal filter in a 2.5-in.-diam by 1-ft-high canister, heated by 2 kw of electrical heat.

51. Waste Tank for Fertile Salt

Purpose: Short-term storage for fertile salt waste.

Design Basis:

Temperature: 600°C

Capacity: 10 days' retention

Material of construction: INOR-8

Description: A tank, 14 in. in diameter and 2.5 ft high, with 14 kw of electrical heat.

52. Makeup Tank for LiF-ThF₄

Purpose: For blanket-salt makeup in the nonshielded operating area to supply the main makeup tanks.

Design Basis:

Temperature: 600°C

Volume: 1 ft³

Material of construction: INOR-8

Description: A tank, 1 ft in diameter and 1.5 ft high, with 7.5 kw of electrical heat.

53. Makeup Tank for Aqueous Flush

Purpose: For makeup and supply of an aqueous flush for decontamination.

Design Basis:

Temperature: Ambient

Volume: 62 ft³

Material of construction: Stainless steel

Description: A tank, 3.5 ft in diameter and 6.5 ft high.

54. Line Heater

Purpose: To heat sections of process lines for transferring molten salt.

Design Basis:

Type: Clam-shell electrical

Rating: 5 kw per 10 ft of 1/2-in. line

Description: Heaters furnished in 10-ft sections.

55. Cooling Air Blower

Purpose: To supply cooling air where needed in the process.

Design Basis:

Capacity: 10,000 ft³/min

Pressure drop: 10 in. H₂O

Material of construction: Steel

Description: Radial-flow blower with a capacity of 10,000 ft³/min at 12 in. H₂O.

PLANT DESIGN AND LAYOUT

The inherent advantage of a fluid-fueled reactor is in close-coupling the reactor and processing plants, thereby reducing capital investment, inventory, and operating costs. This conceptual MSBR processing plant is located in two cells adjacent to the reactor shield; one contains the high-radiation-level operations, and the other contains the lower-radiation-level operations. Each cell is designed for top access through a removable biological shield having a thickness equivalent to 6 ft of high-density concrete. Both cells are served by a crane used in common with the reactor plant. Process equipment is located in the cell for remote removal and replacement from above. No access into the cells will be required; however, it is possible with proper decontamination to allow limited access into the lower-radiation-level cell. A general plan of the processing plant and a partial view of the reactor system is shown in Dwg. 58050D in Appendix F.

The highly radioactive cell contains only fuel-stream processing equipment: the fluorinator, still, waste receiver, NaF and MgF₂ sorbers, and associated vessels. The other cell houses the blanket-processing equipment, fuel- and fertile-stream cold traps, UF₆ reduction equipment, and fuel- and fertile-stream makeup vessels.

The high-radiation-level cell has a cross section of about 19 by 22 ft; the less radioactive cell measures about 27 by 30 ft. Each cell is about 40 ft high, the same as that of the adjoining reactor cell. A "cold" operating area, located along the face of the cells, has dimensions 12 by 62 ft and contains the cold makeup equipment, product receivers, and process-gas supply. All process operations are carried out remotely from the reactor control room.

COST ESTIMATE

One of the major goals in this design study was to estimate the cost of an integrated facility for processing the fuel and blanket salts for this conceptual 1000-Mw (electrical) MSBR. The estimate arrived at includes the total fixed capital cost and the annual operating cost which are, respectively, \$5,301,510 and \$787,790 per year. In general, conventional estimating practices were used except where past experience in the nuclear energy field indicated changes.²⁹⁻³²

It is difficult to separate the process building and its equipment from the total reactor facility for cost-estimating purposes, since the building must be an integral part of the reactor installation. To determine the relationship between the costs of integrated processing plant and the reactor building, it was assumed that the Molten Salt Converter Reactor (MSCR) facility, designed by Sargent and Lundy Engineers,³³ represented the nonprocessing part of an MSBR. Any required addition to the structure, facility, or operating cost of the MSCR due to the addition of the processing facility was considered a cost of the processing facility.

Process Equipment

The installation charge for a component to be installed in an indirect-maintenance area was set at 50% of the selling price and at 25% for equipment to be installed in the direct-maintenance and cold areas. A list of all major installed components and their costs is given in Table 3. The total cost of the installation of major process equipment is \$853,730.

Table 3. Installed Cost of Major Process Equipment

Equipment Number From Fig. 12	Description	No. of Units	Installed Cost
1	Flow control	15	\$ 3,160
2	Cooling tank	1	184,000
3	Fluorinator	1	5,420
4	Surge tank	1	139,000
5	Movable-bed sorber	2	21,400
6	NaF supply tank	1	2,750
7	NaF waste and cooling station	9	12,750
8	Fission-product trap	1	820
9	Cold trap and UF ₆ vaporizer	3	31,930
10	Refrigeration unit	1	6,300
11	Cold trap	4	12,280
12	Refrigeration unit	1	5,000
13	UF ₆ trap	2	420
14	Fluorine surge tank	1	1,910
15	Fluorine compressor	1	1,850
16	Distillation unit	1	58,700
17	Makeup tank	1	1,570
18	Makeup tank	1	4,390
19	Cooling tank	1	58,000
20	Makeup tank	1	6,590
21	Vacuum pump	1	2,640
22	Reduction column	1	9,050
23	Molten-salt pump	1	15,000
24	Collection tanks	2	11,600
25	Filters	2	900
26	Scrubbing column	1	1,240
27	Fission-product hydrator	1	400
28	Condenser	1	800
29	Absolute filter	1	200
30	Makeup tank	1	1,410
31	Pump	1	200
32	Waste tank	1	4,540
33	Fluorine storage and supply	2	13,800
34	Hydrogen storage and supply	4	780
35	HF storage and supply	2	1,280
36	NaK cooler	1	60,500
37	NaK heater	1	38,670
38	Collection tank	1	6,380
39	NaK pump	1	2,000
40	Supply tank	1	6,670
41	Supply tank	1	3,430
42	Fluorinator	1	16,600
43	Movable-bed sorber	1	2,120
44	Fission-product trap	1	430

Table 3 (continued)

Equipment Number From Fig. 12	Description	No. of Units	Installed Cost
45	Refrigeration unit	1	3,600
46	Refrigeration unit	1	3,300
47	Product receiver	1	200
48	UF ₆ trap	1	250
49	Makeup tank	3	63,300
50	Molten-salt filter	4	1,060
51	Waste tank	1	2,940
52	Makeup tank	1	1,570
53	Aqueous-flush makeup tank	1	7,400
54	Line heaters	13	6,500
55	Cooling-air blower	2	4,760
Total installed equipment cost			\$853,760

Several of the process vessels and auxiliary equipment are similar to equipment previously purchased by ORNL or previously estimated in another design report.²⁸ These estimates were updated, and in some cases conventional equipment sizes were extrapolated to meet the requirements of this larger plant. The process vessels in which high heat fluxes were anticipated were considered to have costs comparable to that of the MSRE drain-and-fill tank.

For vessels and tanks of conventional design, the cost was computed from the cost of material plus estimated charges for fabrication and installation. A summary of the fabrication charges used is given in Table 4.

Additional estimating criteria were:

Vessel-wall thickness: 1/2-in. or sched-40 pipe

Vessel-end thickness: <36-in. diam, 1/2 in.

>36-in. diam, 3/4 in.

Equivalent vessel weight for entrances and exits: <100 lb vessel, 10 lb

100-1000 lb vessel, 20 lb

>1000 lb vessel, 40 lb

Cost of electrical heaters: \$100/kw installed

Table 4. Summary of Fabrication Cost for Process Vessels^{29,32}

Material	Fabrication Cost (\$/lb) for a Vessel Weight of:	
	<1000 lb	>1000 lb
Mild steel	2.00	1.00
Stainless steel, nickel, Monel, and Inconel	3.50	2.50
INOR-8	4.50	3.50
Alloy 79-4	5.00	4.00

Structure and Improvements

Structure and improvements costs for the processing facility were determined by assuming that these costs are directly related to corresponding costs developed for the MSCR by Sargent and Lundy Engineers.³³ Addition of the processing facility proposed here called for an additional 16.56% of floor space in the reactor building, and this factor was used to determine the cost from the corresponding MSCR data. The cost of the crane and hoist was also increased by this amount. This additional space includes a 10% increase in analytical and decontamination facilities. The processing facility addition was 7.3% of the total building area; therefore, cost of the grounds and stack was increased by 7.3%. The total increase in cost of structure and improvements due to the processing facility was \$556,770, which was obtained as follows:

Structure	0.1656 x \$2,932,400 = \$485,610
Crane and hoist	0.1656 x 195,000 = 32,290
Grounds	0.073 x 501,500 = 36,610
Stack	0.073 x 31,000 = <u>2,260</u>
Total structures and improvements	\$556,770

Interim Waste Storage

The interim-waste-facility cost was estimated separately. This estimate was based on previous cost estimates for similar facilities.^{14,32}

Total waste facility cost was \$387,970, with the following cost breakdown:

Tank cost	\$213,200
Excavation and backfill	32,350
Concrete	18,070
Transfer lines	9,000
Instrumentation	10,000
Cooling system	100,650
Structural steel	4,700
Total	\$387,970

Other Plant Costs

Significant cost items in the indirect-maintenance area of the plant are the process line "jumpers", which have remote connectors necessary for indirect connection of pipelines, instrumentation lines, and electrical lines. The following cost schedule, based on experience at ORNL, was used for estimating the cost of these connections:³²

Major pipelines	\$1,500/line
Multiple pipe and instrumentation lines	\$1,700/set
Electrical heater connection, including the heater	\$2,000/set

Other process-piping cost schedules were:

Motor-operated control valves	\$500 each
Coolant air ducts	\$10/ft
Major process lines in direct maintenance area (<20 ft)	\$400 each
Gas lines in direct maintenance area (<20 ft)	\$200 each
NaK coolant lines (<20 ft)	\$800 each

The above cost schedule results in a total process-piping cost of \$155,800.

The electrical auxiliaries consisted of the electrical substation, switching gear, feeders, and indirect connectors and jumpers. Cost schedules used for these auxiliaries were:²⁹

Electrical substation and switching	\$36.60/kw
Overhead feeders	\$6.1/ft
Underground feeders	\$12.2/ft

The total cost of electrical auxiliaries is estimated to be \$84,300.

Process instrumentation is estimated to be \$272,100, radiation monitoring to be \$100,000, and sampling connections to be \$20,000.

The costs of service lines and high-temperature insulation are based on the installed cost of the process equipment in the main processing

areas and in the waste storage facility. The cost of service lines, taken at 15% of the total installed cost of this equipment, amounts to \$128,060; the cost of the insulation, taken at 6% of this installed cost, amounts to \$51,220.

Total Fixed Capital Cost

The total direct plant costs are estimated to be \$2,609,980 (Table 5). Past experience was used to determine percentage costs of indirect capital items. These percentages are higher than those for other chemical industries but represent actual cost experience in other ORNL projects.³² Construction overhead is estimated at 30% of total direct plant cost to give a total construction cost of \$3,392,970. Engineering and inspection at 25% of total construction cost is \$848,240, which results in a subtotal plant cost of \$4,241,210. Contingency at 25% of the subtotal plant cost is \$1,060,300, and the total construction cost is \$5,301,510.

Inventory costs include the cost of the molten salt held up in the system and the initial cost of the NaK coolant. Total fuel-salt holdup for this system is 63 ft³. At \$1,420/ft³, the charge is \$89,460. Blanket-salt holdup is 120 ft³, and at \$560/ft³ the cost is \$67,200. Fuel- and blanket-salt charges do not include the cost of fissile material. The 400-ft³ holdup of NaK at \$100/ft³ costs \$40,000. The total inventory cost is then \$196,660, giving a total fixed capital cost of \$5,498,170.

Direct Operating Cost

The direct operating cost includes the cost of operators, chemical consumption, waste containers, utilities, and maintenance materials. The number of operating and support employees is based on a work schedule of four shifts of 40 hr each per week. These include immediate supervisory, operating, maintenance, laboratory, and health physics personnel, plus two people for routine clerical and janitorial work (Table 6).

No attempt was made to prorate the cost of higher supervisory, clerical, or plant protection personnel for the processing facility since some of this cost is included in labor overhead costs.

Table 5. Total Fixed Capital Cost

Installed process equipment	\$ 853,760
Structure and improvements	556,770
Interim waste storage	387,970
Process piping	155,800
Process instrumentation	272,100
Electrical auxiliaries	84,300
Sampling connections	20,000
Utilities (15% of installed process equipment)	128,060
Insulation (6% of installed process equipment)	51,220
Radiation monitoring	<u>100,000</u>
Total direct plant cost	\$2,609,980
Construction overhead (30% of total direct plant cost)	<u>782,990</u>
Total construction cost	\$3,392,970
Engineering and inspection (25% of total construction cost)	<u>848,240</u>
Subtotal plant cost	\$4,241,210
Contingency (25% of subtotal plant cost)	<u>1,060,300</u>
Total construction cost	\$5,301,510
Inventory cost	
Molten fuel salt (at \$1,470/ft ³)	89,460
Molten blanket salt (at \$560/ft ³)	69,200
NaK (at \$100/ft ³)	<u>40,000</u>
Total inventory cost	\$ 196,660
Total fixed capital cost	\$5,498,170

Table 6. Employment Costs

	No.	Cost (\$/year)
Production		
Shift supervisor	4	32,000
Operators	16	96,000
Maintenance workers	8	48,000
Laboratory analysts	4	24,000
Health physics workers	2	12,000
Others	2	<u>10,000</u>
Total	36	222,000

The cost of utilities, waste containers, and consumed chemicals is based on a 300 day/year operation for both the reactor and processing plant.

Total direct operating cost for one year is \$610,190 (Table 7); this includes fuel and fertile salt makeup.

Processing Cost

The processing cost per year is estimated at \$1,447,570 (Table 8). This cost is obtained by combining the direct operating cost, the indirect cost of labor overhead (80% of direct labor cost), the fixed cost due to depreciation (10% of fixed capital per year), taxes (1% of fixed capital per year), and insurance (1% of fixed capital per year). The percentage used for the indirect labor cost is arbitrary; however, it is within the range of usual practice.

On the basis of 300 days of operation per year for the 1000-Mw (electrical) MSBR, the fuel-processing cost is 0.201 mill/kwhr. The fuel-cycle cost is composed of this cost plus the in-reactor inventory of fuel, fertile, and carrier salts, plus makeup fertile and carrier salts, and less the credit for excess $^{233}\text{UF}_6$ produced. In-reactor inventory, makeup, and credit were not considered in this study.

Table 7. Annual Direct Operating Cost

Labor		\$222,000
Chemical consumption		
Fluorine (at \$2.00/lb)	\$ 4,080	
KOH (at \$0.10/lb)	980	
Hydrogen (at \$0.01/ft ³)	720	
HF (at \$0.26/lb)	1,000	
NaF pellets (at \$1.00/lb)	5,780	
MgF ₂ pellets (at \$1.00/lb)	420	
Inert gases (guess)	830	
Fuel-salt makeup (at \$1,420/ft ³)	28,400	
Blanket-salt makeup (at \$560/ft ³)	27,350	
NaK makeup (guess)	<u>830</u>	70,390
Waste containers		28,270
Utilities		
Electricity (at \$0.01/kwhr)	73,300	
Others (guess)	<u>7,000</u>	80,300
Maintenance materials		
Site (guess)	2,500	
Building (at 2% of building cost)	10,810	
Service and utilities (at 4% of service and utilities cost)	35,880	
Process equipment (at 15% of process equipment cost)	<u>160,040</u>	<u>209,230</u>
Total annual direct operating cost		\$610,190

Table 8. Annual Fuel-Processing Cost

Direct operating cost	\$ 610,190
Labor overhead (80% of labor cost)	177,600
Fixed costs	
Depreciation (10%/year of fixed capital)	549,820
Tax (1% of fixed capital)	54,980
Insurance (1% of fixed capital)	<u>54,980</u>
Total	\$1,447,570

Within the applied ground rules, these costs are believed to be a reasonably accurate representation of the cost for regenerating the fuel and blanket in an integrated MSBR processing plant. A more thorough study would have included detailed design of equipment and layout of the integrated processing plant, the reactor and its auxiliaries. Such a thorough analysis was beyond the scope of this study.

CONCLUSIONS AND RECOMMENDATIONS

The central issues in this preliminary study were to analyze the feasibility and cost of a conceptual system for continuously regenerating the fuel and fertile streams in the Molten Salt Breeder Reactor. Briefly, the system consists in (1) fluorinating, distilling, and reconstituting the molten fluorides used in the reactor core, and (2) recovering the ^{233}U from the molten breeder blanket by fluorination and using the uranium to reconstitute the core salt. The excess is to be sold. The power of the breeder reactor was set at 1,000 Mw (electrical) for this study.

A number of basic conclusions and essential recommendations were developed. The conclusions relate to the projected feasibility and anticipated costs in terms of established technology and cost-accounting practices, and the recommendations refer to what must yet be learned with respect to technology and chemical data before a complete engineering analysis can be made. It is our opinion that it will be very useful to

begin filling these gaps in the knowledge because of the promising simplicity and low cost shown by this study.

In the conclusions and recommendations presented below, the recommendations are underlined. The first four numbered paragraphs relate to general characteristics of this fuel-processing plant, and the others to specific unit operations.

1. FEASIBILITY. Fluorination followed by distillation is a feasible process for regenerating MSBR fuel ($\text{LiF}-\text{BeF}_2-\text{UF}_4$). Fluorination alone is sufficient processing for the fertile salt ($\text{LiF}-\text{ThF}_4$). Reactor physics calculations indicate that attractive breeding ratios can be obtained for such a process. Engineering problems in the processing plant appear to be amenable to solution through a well-planned developmental program at the unit-operations level. Fluorination-distillation should be developed as the processing method for the MSBR. Concurrently, other attractive processes should be investigated at the laboratory and/or engineering stage as potential alternatives.
2. INTEGRATED PLANT. Integrating processing and reactor facilities is of primary importance in lowering the processing cost. Complete advantage is thereby taken of the ready adaptability of a fluid-fueled reactor for continuous processing with corresponding minimum inventory. The relatively small size of this side-stream processing plant, about 12 ft^3 salt per day for a 1000-Mw. (electrical) reactor, is amenable to integrated construction, thereby separating the economic dependence of the processing industry upon a large amount of installed electrical capacity. The same financing convention that applies to the power plant applies to the processing plant; this type of financing is normally available at a lower rate than is available for a separated processing facility.
3. ECONOMY. The estimated capital cost, excluding inventory, of the plant is \$5,301,000, and operating costs are about \$788,000 a year. It is significant that the capital investment in the integrated processing plant proposed here is only about 4% of the total cost of the reactor system it serves.
4. CORROSION. There are at least three areas in the chemical processing plant in which corrosion behavior of construction materials should be studied. These are the vacuum still, the reduction unit, and filters. The still temperature of about 1000°C is much higher than has been contemplated for any other part of the MSBR system, and the resistance of INOR-8 and nickel to corrosive attack at this temperature is not known. A reducer, hydrogen, enters the reduction unit and probably helps limit corrosion there, but this should be verified. Filters are subject to attack because of the large surface area exposed to the fluid being filtered.

5. FLUORINATION DEVELOPMENT. Batchwise fluorination of molten fluoride salts for uranium recovery has been rather thoroughly investigated at ORNL; however, it is recommended that engineering development of a continuous fluorinator be given high priority. The need for continuous fluorination is evidenced by the requirement of low fuel and carrier salt inventories in the processing plant. The reactor fuel system contains about 650 ft³ of salt, and, without continuous fluorination, the out-of-pile inventory could possibly be as large as the in-pile inventory. This study indicates that in a continuous fluorination-distillation process the holdup represents about 10% of the reactor fuel volume.

It is recommended that the frozen-wall concept for a continuous fluorinator be developed and demonstrated. This concept calls for a 1/2- to 3/4-in.-thick layer of frozen salt on the inside wall of the fluorinator to prevent the serious corrosive attack by the molten mixture during fluorination. Basic information needed includes fluorination rate data, process control in continuous operation, and method of establishing and maintaining a frozen wall.

Fluorination of the fertile salt introduces problems similar to those encountered for the fuel stream. However, fertile-stream processing rates are 8 to 10 times higher, and the fission-product activity is several orders of magnitude less. On the other hand, a lower fluorination efficiency can be tolerated in blanket processing.

Another method for continuous fluorination is the gas-phase continuous operation in which fluoride microspheres are generated and fluorinated as they fall through a tower. This process should be recognized as a potential alternative to the continuous method of fluorination studied here, but its development should be subordinate to that of the frozen-wall concept.

6. DISTILLATION DEVELOPMENT. The vacuum-distillation concept for separating the LiF-BeF₂ (fuel-carrier salt) from fission products is feasible from an engineering viewpoint. The theoretical net discard of ⁷Li in the still waste is low enough that its cost is insignificant compared with other fuel cycle costs. Thermal problems require that sufficient volume be maintained for wetting a large heat transfer surface, and the buildup of fission products in this volume will almost surely have an adverse effect on the decontamination factor of the distillate. Relative volatility data are needed for the multicomponent mixture, LiF-BeF₂-fission products, in which compositions are in the range 99.5-0.5 mol % to 84.5-0.5-15 mol % LiF-BeF₂-rare earth fluorides. It is strongly recommended that a continuous vacuum still be built and operated to demonstrate a workable design and to obtain rate and entrainment data.

7. REDUCTION OF UF_6 TO UF_4 . This reaction is quantitative when UF_6 is fed into an $\text{H}_2\text{-F}_2$ flame containing excess H_2 , producing a powdery UF_4 product. The solid product cakes and adheres to vessel walls, which should be avoided if possible in a remotely operated system. The liquid-phase reaction proposed here is more suitable for remote operation and should be developed. The operating conditions that need study are: temperature, UF_6 concentration, reaction rate, nozzle design, circulation rate, contactor design, and gas-liquid separation.

The reducing conditions that exist in this operation are consistent with those required for purging nickel, chromium, and iron corrosion products from the fuel. Therefore the potentiality of using the reduction unit for simultaneously giving the fuel a final cleanup should be investigated.

8. SOLID-LIQUID SEPARATION. The general area of high-temperature, solid-liquid separation in remote operations needs development. Filtration techniques should surely be investigated to determine operability and reliability in molten-fluoride systems.
9. FISSION PRODUCT BEHAVIOR. A better understanding of fission product behavior throughout the processing plant is needed. In particular, data are needed on the ways in which the various nuclides partition in the several processing steps and on the efficiency of removal. A more basic study is concerned with the behavior of the fission products in the reactor environment to determine whether or not certain nuclides remain in the reactor system.
10. URANIUM HEXAFLUORIDE PURIFICATION. The NaF and MgF_2 sorption units provide adequate decontamination for UF_6 . The batchwise units can be operated satisfactorily for both fuel and fertile streams of the MSBR; however, a continuous, temperature-zoned system would reduce the frequency of cell entry. Probably the largest uncertainties in UF_6 purification are in the removal of tellurium and ruthenium; means of removing them should be developed.

The portion of the fertile stream used as fuel makeup need not be passed through a sorption system because of the very small amount of fission-product contamination. However, the UF_6 product that would eventually be handled by contact would need purification.

11. PROTACTINIUM REMOVAL. The most significant advancement in fertile-stream processing can be made in the development of a process that removes protactinium. To be effective, the process must remove protactinium five to ten times as fast as its decay rate; that is, the blanket volume would have to be processed every four to eight days. Simplicity and ease of operation are obvious requirements. Thus, a process based on forming protactinium oxide by ion exchange appears promising and should be studied.

Effective removal would obviate the need of sending any portion of the fertile stream to waste and possibly reduce makeup cost to that represented by thorium burnup alone. Such a process is not required for economic operation of the MSBR since fluorination and fractional discard can adequately control the fission product concentration; however, the potential advantage of reduced waste cost and improved breeding performance argues for basic development of a process.

12. FUEL-STREAM WASTE SYSTEM. The most economical method of waste management consists in bulk storage in a large heat exchanger tank. The heat generation problem is so severe that the plant waste must be diluted with an inexpensive, inert material, which complicates future processing for recovery of any contained values. The calculated loss of ^7Li and ^{233}U in the fluorination-distillation process is small, being only 1.5 to 2% of the fuel-cycle cost.

The possible use of fission-product decay heat should not be overlooked in the evaluation of an MSBR. The accumulated waste generates about 4.5 Mw (thermal) throughout most of the filling period of the waste tank. Thus salt-storage temperature can be maintained high enough to make the waste tank a source of high-temperature energy.

13. FERTILE-STREAM WASTE SYSTEM. The fertile-stream waste contains a significant inventory of valuable materials whose recovery is probably warranted. At the end of the 30-year filling period, the waste tank contains about 141,000 kg of Th, 10,400 kg of ^7Li , and 116 kg of ^{233}U ; the ^{233}U is isotopically pure material, having been formed almost entirely out of the fission zone. Although not considered in this study, an in-cell decay period of about six months followed by refluorination appears to be advisable for greatly reducing the amount of ^{233}Pa that enters the waste tank.
14. PROCESS CONTROL. This aspect of plant operation was given only a cursory review in this study, and no areas of unusual control difficulty were observed. A flow-control device for the molten-salt stream is needed, and the dynamic freeze-valve concept should be developed. Analytical and sampling requirements require a more thorough study than was given here.

ACKNOWLEDGEMENT

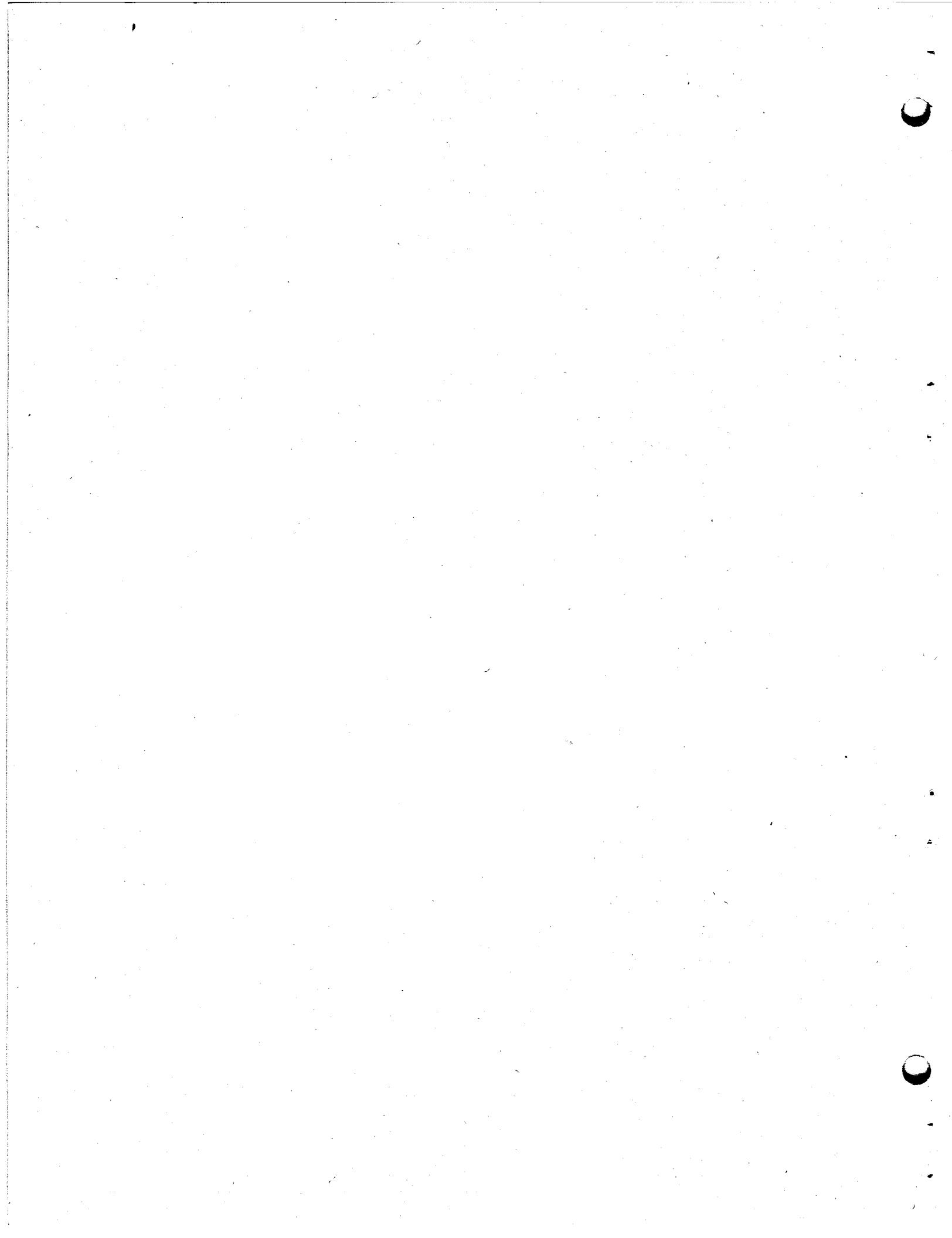
W. G. Stockdale assisted the authors in the development of the cost of the plant and equipment, and his help is gratefully acknowledged.

REFERENCES

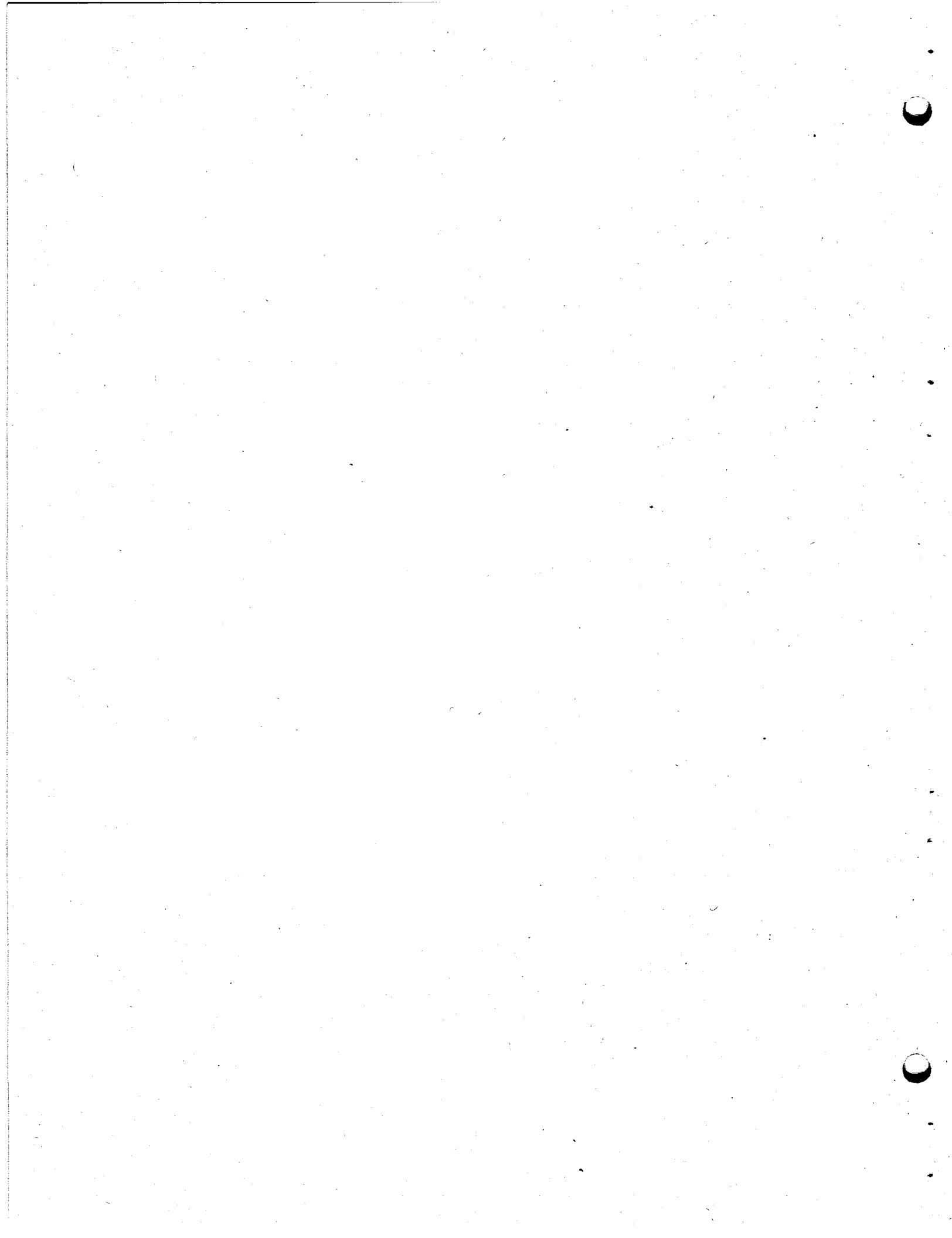
1. E. S. Bettis, ORNL, personal communication.
2. H. F. Bauman and E. S. Bettis, ORNL, personal communication on unpublished work still in progress.
3. M. J. Kelly, ORNL, personal communication on unpublished work still in progress.
4. E. D. Arnold, ORNL, personal communication on unpublished work.
5. R. E. Thoma, Rare Earth Halides, ORNL-3804 (May 1965), p. 42.
6. C. B. Jackson (ed.), Liquid Metals Handbook, Sodium (NaK) Supplement, 3d ed., U.S.A.E.C., Washington, D.C., 1955.
7. K. A. Sense, M. J. Snyder, and J. W. Clegg, "The Vapor Pressure of Beryllium Fluoride," *J. Phys. Chem.* 58, 223-4 (1954).
8. O. Ruff, G. Schmidt, and S. Mugdan, "Arbeiten aus dem Gebiet Hoher Temperatur. XV. Die Dampfdrucke der Alkalifluoride," *Z. Anorg. Allgem. Chem.* 123, 83-8 (1922).
9. H. von Wartenberg and H. Schulz, "Der Dampfdrucke Einiger Salze. II, Z. fur Elektrochemie" 27, 568-73 (1921).
10. K. A. Sense and R. W. Stone, "Vapor Pressures and Molecular Compositions of Vapors of the RbF-ZrF₄ and LiF-ZrF₄ Systems," *J. Phys. Chem.* 62, 1411-8 (1958).
11. K. A. Sense and R. W. Stone, "Vapor Pressures and Molecular Compositions of Vapors of the Sodium Fluoride-Beryllium Fluoride System," *J. Phys. Chem.* 62, 453-7 (1958).
12. M. M. Popov, F. A. Kostylev, and N. V. Zubova, "Vapor Pressures of Uranium Tetrafluoride," *Zhur. Neorg. Khim.* 4, 1708-9 (1959).
13. W. R. Grimes, Reactor Chemistry Division Annual Progress Report for Period Ending January 31, 1960, ORNL-2931 (Apr. 29, 1960), p. 50-1.
14. W. L. Carter and J. B. Ruch, A Cost Estimate for Bulk Disposal of Radioactive Waste Salt from Processing Zirconium-Uranium Fuel by the ORNL Fluoride Volatility Process, ORNL-TM-948 (Sept. 25, 1964) (Confidential).
15. D. O. Campbell and G. I. Cathers, ORNL, personal communication of unpublished data.

16. G. Burrows, Molecular Distillation, p. 18-31, Oxford University Press, Great Britain, 1960.
17. W. H. Carr, S. Mann, and E. C. Moncrief, "Uranium-Zirconium Alloy Fuel Processing in the ORNL Volatility Pilot Plant," paper presented at the A.I.Ch.E. Symposium, "Volatility Reprocessing of Nuclear Reactor Fuels," New York, N. Y. (Dec. 1961).
18. R. P. Milford, S. Mann, J. B. Ruch, and W. H. Carr, Jr., "Recovering Uranium Submarine Reactor Fuels," Ind. Eng. Chem. 53, 357 (1961).
19. J. C. Mailen, "Volatilization of Uranium as the Hexafluoride from Drops of Molten Fluoride Salt," paper presented at the A.C.S. National Meeting in Chicago, Ill. (Sept. 2, 1964).
20. G. I. Cathers, M. R. Bennett, and R. L. Jolly, The Fused Salt-Fluoride Volatility Process for Recovering Uranium, ORNL-2661 (1959).
21. M. E. Whatley et al., Unit Operations Section Monthly Progress Report, September 1963, ORNL-TM-785 (1964).
22. E. C. Moncrief, Results of Volatility Pilot Plant Cold Uranium Flowsheet Demonstration Runs TU-6 and TU-7, ORNL-CF-61-9-37 (1961).
23. W. W. Pitt, unpublished data, ORNL (1965).
24. R. W. Kessie et al., Process Vessel Design for Frozen Wall Containment of Fused Salt, ANL-6377 (1961).
25. M. J. Kelley, ORNL, personal communication (1964).
26. L. E. McNeese and C. D. Scott, Reconstitution of MSR Fuel by Reducing UF₆ Gas to UF₄ in a Molten Salt, ORNL-TM-1051 (1965).
27. C. D. Scott, unpublished data (1965).
28. W. L. Carter, R. P. Milford, and W. G. Stockdale, Design Studies and Cost Estimates of Two Fluoride Volatility Plants, ORNL-TM-522 (1962).
29. John Happel, Chemical Process Economics, Wiley, New York, 1958.
30. R. S. Aries and R. D. Newton, Chemical Engineering Cost Estimation, McGraw-Hill, New York, 1955.
31. C. H. Chilton (ed.), Cost Engineering in the Process Industries, McGraw-Hill, New York, 1960.
32. W. G. Stockdale, ORNL, personal communication (1965).
33. Sargent and Lundy Engineers, Capital Investment for 1000-Mwe Molten Salt Converter Reference Design Power Reactor, SL-1994 (1962).

34. Octave Levenspiel, "Chemical Reaction Engineering," Wiley, New York, 1962.
35. W. R. Gambill, "Fused Salt Thermal Conductivity," Chem. Eng. 66 (16), 129-30 (1959).
36. B. C. Blanke et al., Density and Viscosity of Fused Mixtures of Lithium, Beryllium, and Uranium Fluorides, Mound Laboratory Report MLM-1086 (Mar. 23, 1959).
37. S. I. Cohen, W. D. Powers, and N. D. Greene, A Physical Property Summary for ANP Fluoride Mixtures, ORNL-2150 (Aug. 23, 1956).
38. W. R. Grimes, Reactor Chemistry Annual Progress Report for Period Ending January 31, 1963, ORNL-3417, p. 47.
39. M. E. Lackey, Internally Cooled Molten-Salt Reactors, ORNL-CF-59-6-89 (1959).



APPENDICES



APPENDIX A. DESIGN CALCULATIONS FOR FUEL SALT FLUORINATOR AND COOLING TANK

The unit operation of fluorination requires temperature control at about 550°C for the salt being fluorinated. The fact that this salt is highly radioactive introduces a problem because it is necessary to extract this decay heat through the walls of a fluorinator whose size is fixed by other process requirements, such as throughput and residence time. If there is insufficient heat transfer surface available for this purpose, then the salt must be allowed to "cool" before entering the fluorinator. The solution to this problem is to insert a cooling tank immediately upstream of the fluorinator.

The following calculations pertain to the thermal design and size of the fuel-salt fluorinator and the size of the cooling tank. It was determined that the maximum permissible heat flux for the fluorinator is $5.31 \times 10^4 \text{ Btu hr}^{-1} \text{ ft}^{-3}$ and that the size is 4.75 in. in diameter by 10.3 ft high. The cooling tank requires a volume of 22.5 ft³. A further result of this calculation is a graph of heat generation rate at the cooling tank exit as a function of elapsed time since discharge from the reactor.

The fluorinator design criteria are:

Fuel-salt flow rate	15 ft ³ /day
Uranium rate	77.3 g-moles/day
Fluorine utilization	33-1/3%
Maximum mass flow rate of gas	0.277 slpm/in. ²
Maximum heat flux through the frozen wall	1.5 kw/ft ²
Heat of reaction ($\text{UF}_4 + \text{F}_2 = \text{UF}_6$)	-162 kcal/g-mole
Residence time of salt	2 hr

The F_2 flow rate through the column is:

$$\begin{aligned} F &= 77.3 \times 3 = 231.9 \text{ g-moles}, \\ &\quad = 4.83 \text{ slpm}. \end{aligned}$$

Fluorinator cross-sectional area is:

$$A = \frac{4.83}{0.277} = 17.5 \text{ in.}^2,$$

or the column diameter is:

$$D = \sqrt{\frac{4 \times 17.5}{\pi}} = 4.75 \text{ in.}$$

The total column volume needed is:

$$V = \frac{15 \times 2}{24} = 1.25 \text{ ft}^3 .$$

Therefore, the column height is:

$$h = \frac{1.25 \times 144}{17.5} = 10.3 \text{ ft} .$$

The area of the frozen wall per linear foot of column is:

$$A_F = \frac{\pi \times 4.75}{12} = 1.24 \text{ ft}^2 .$$

The open volume per foot of column is:

$$V_o = 0.1204 \text{ ft}^3 .$$

Therefore, the maximum heat removal rate in the frozen-wall column is:

$$H = \frac{1.25 \times 1.5 \times 3413}{0.1204} = 5.31 \times 10^4 \text{ Btu hr}^{-1} \text{ ft}^{-3} .$$

But, the heat of reaction contributes at the following rate:

$$\frac{77.3 \times 162}{24 \times 1.25 \times 0.252} = 1620 \text{ Btu hr}^{-1} \text{ ft}^{-3} .$$

Thus, allowable fission product heating is:

$$5.15 \times 10^4 \text{ Btu hr}^{-1} \text{ ft}^{-3} .$$

The surge tank must have sufficient capacity to allow the fuel salt to cool to a heat rate of $5.15 \times 10^4 \text{ Btu hr}^{-1} \text{ ft}^{-3}$. If the tank is assumed to be well-mixed, and assuming that the heat generation rate of the salt can be expressed as an exponential function of time, then,³⁴

$$H(t) = Ke^{-kt} \text{ (heat generation rate)}, \quad (A-1)$$

$$E(t) = \frac{1}{\tau} e^{-t/\tau} \text{ (age distribution function)}, \quad (A-2)$$

where

τ = V/F , average residence time,

V = fluid volume in tank,

F = volumetric flow rate,

t = time since exit from reactor core,

K, k = constants.

The heat generation rate of molten salt from the exit of the well-mixed surge tank can be expressed as:

$$H_E = \int_0^\infty E(t) H(t) dt . \quad (A-3)$$

The data on heat generation rate with respect to time elapsed since the salt has been removed from the reactor can be expressed as a series of equations of the form of Eq. (A-1). Six intervals were chosen from the data in Figs. 2a and b (see text). In each time interval, the constants of the exponential equation approximation were determined by coupling the equation at the two ends of the time interval. This resulted in a representation of the heat generation data, which was always equal to or somewhat greater than the calculated heat generation rate. The values of the constants in the approximate equations are:

No.	Time Interval (hr)	K	k
1	0-0.0167	5.98×10^5	24.2
2	0.0167-0.167	4.28×10^5	3.98
3	0.167-1	2.46×10^5	0.651
4	1-10	1.41×10^5	0.098
5	10-100	5.85×10^4	0.0104
6	100 →	2.43×10^4	0.00154

After substitution of the constants, integration of Eq. (A-3) by time segments gives:

$$\begin{aligned}
 H_E = & \frac{1}{\tau} \left\{ \frac{5.98 \times 10^5}{(\frac{1}{\tau} + 24.2)} \left[1 - e^{-0.01667 (\frac{1}{\tau} + 24.2)} \right] \right. \\
 & + \frac{4.28 \times 10^5}{(\frac{1}{\tau} + 3.98)} \left[e^{-0.01667 (\frac{1}{\tau} + 3.98)} - e^{-0.1667 (\frac{1}{\tau} + 3.98)} \right] \\
 & + \frac{2.46 \times 10^5}{(\frac{1}{\tau} + 0.651)} \left[e^{-0.1667 (\frac{1}{\tau} + 0.651)} - e^{-(\frac{1}{\tau} + 0.651)} \right] \\
 & + \frac{1.41 \times 10^5}{(\frac{1}{\tau} + 0.098)} \left[e^{-(\frac{1}{\tau} + 0.098)} - e^{-10(\frac{1}{\tau} + 0.098)} \right] \\
 & + \frac{5.85 \times 10^4}{(\frac{1}{\tau} + 0.0104)} \left[e^{-10(\frac{1}{\tau} + 0.0104)} - e^{-100(\frac{1}{\tau} + 0.0104)} \right] \\
 & \left. + \frac{2.43 \times 10^4}{(\frac{1}{\tau} + 0.00154)} \left[e^{-100(\frac{1}{\tau} + 0.00154)} \right] \right\} .
 \end{aligned}$$

This equation was used to determine heat generation rate of the molten salt at the cooling-tank exit in terms of average residence time (Fig. A-1). At the design heat generation time, the average residence time was 36 hr. Therefore, the volume of the cooling tank must be:

$$V = \frac{36 \times 15}{24} = 22.5 \text{ ft}^3.$$

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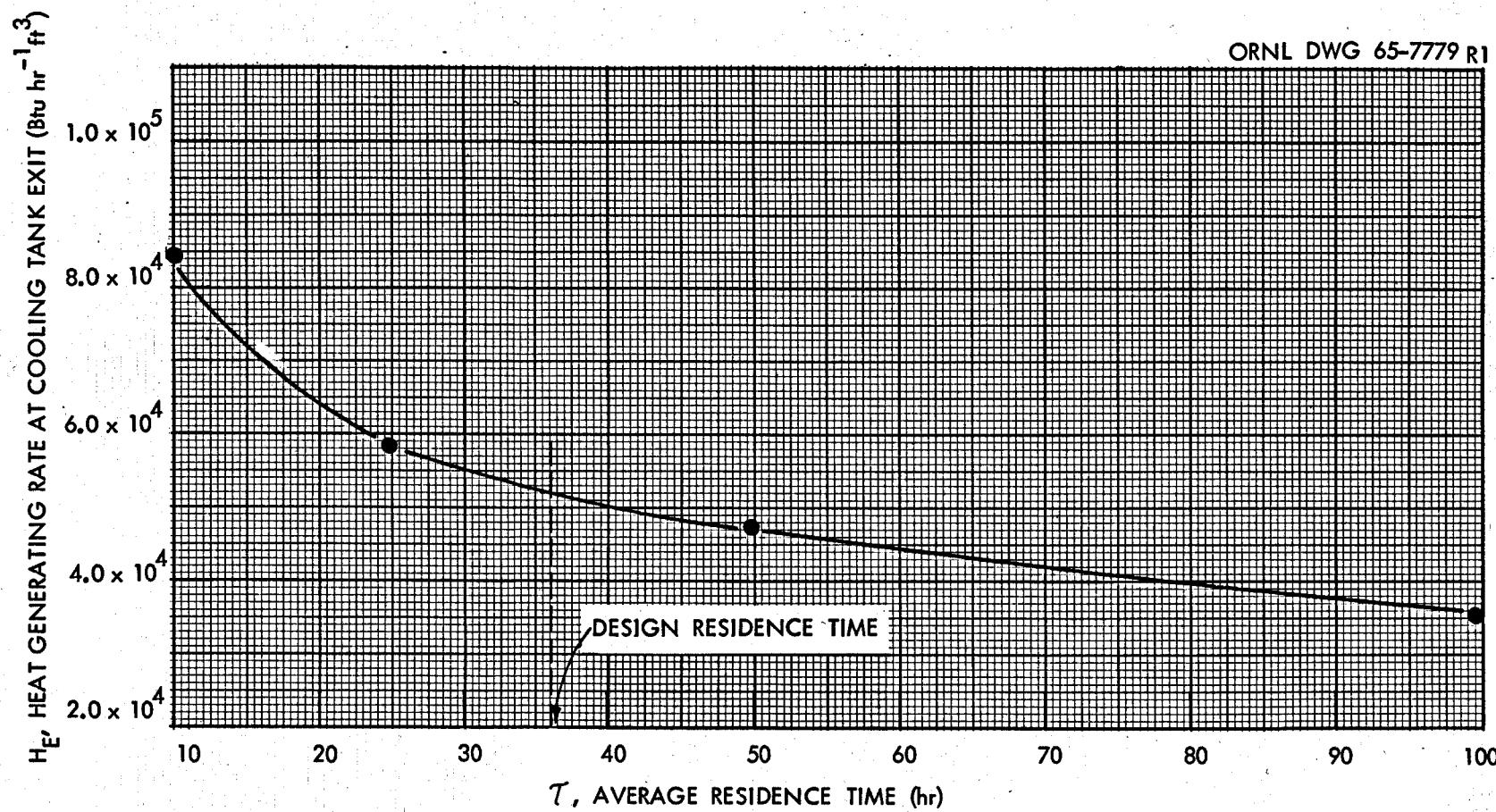


Fig. A-1. Heat Generation Rate of Molten Salt at the Cooling Tank Exit. The salt is given 36 hours of decay time to facilitate temperature control in the fluorinator.

APPENDIX B. FISSION-PRODUCT HEAT GENERATION RATES
IN THE MOBILE-BED SORBERS AND NaF WASTE TANKS

Heat generation rates for the following components are presented here:

- Movable-Bed Sorber
- Sodium Fluoride Waste Tanks
- Short-Term Cooling Station
for Waste Sodium Fluoride
- Interim-Storage of Waste

Fission products, which are volatilized in the fluorinator, accumulate in the movable-bed sorbers and create a heat source that must be considered in the design of these units. Excess heat must be removed so that it does not interfere with control of bed temperature. Actual removal of the heat is not the problem here since sorbers, which accomplish this end, have been designed and used in the ORNL Volatility Pilot Plant.

The essential problem is to estimate the heat generation rate due to radioactive decay of the fission products present in the system projected here. This was done by first assuming that half the fission-product heat generating capacity which reached the fluorinator would exit to the sorber. Further, all the fission-product heat is concentrated in the lower zone in the sorber, and it was removed in the NaF waste stream. As noted before, one-fifth of the lower zone is exhausted per day, and there are two parallel sorbers alternately operating for 12 hr each. This results in an accumulation of decaying fission products or of decreasing heat sources in the sorber, in the NaF waste tanks, and finally in the interim waste-storage facility.

The heat generation rate for various process components was approximated by determining the average heat generation rate during a specific time period, and by assuming that this rate decayed as the total fission-product heat rate decayed as shown in Figs. 2a and b. The accumulated heat generation rate could then be expressed as a rate characteristic of fission products having an "average" age intermediate between the oldest and the most recently sorbed.

Movable-Bed Sorber

In the movable-bed sorber, the heat load can be approximated by assuming that the volatile fission products from the fluorinator accumulate for 5 days and that those first accumulated will decay or cool as additional accumulation occurs. A table can be prepared for accumulated heat generation rate, derived from the residence time of fission products in the sorber:

Residence Time of Fission Products in Sorber (days)	Heat Generation Rate (Btu/hr)
1	144,000
2	120,000
3	106,000
4	96,000
5	<u>86,000</u>
Steady-state heat generation rate in movable-bed sorber	552,000 Btu/hr

Sodium Fluoride Waste Containers

The NaF waste tanks accumulate NaF and fission products from the sorbers. Each tank holds two complete bottom sorber zones from each of two sorbers (10 days' accumulation of fission products in one-day increments). These zones exhaust to the NaF waste tank each day for 10 days. According to the slope of the fission-product decay heat curve (Figs. 2a and b), the average residence time of the fission products, and the average heat generation rate of the NaF bed material as it leaves the sorber, the following heat generation rate exists in the NaF waste tank at the end of 10 days, at which time it is full.

Residence Time of NaF in Container (days)	Average Heat Generation Rate (Btu/hr)
1	110,300
2	83,400
3	71,500
4	62,600
5	56,600
6	53,600
7	50,700
8	47,700
9	41,700
10	<u>38,700</u>

Heat generation rate in NaF
waste tank when filled 616,800 Btu/hr

Short-Term Cooling Station for Waste Sodium Fluoride

The NaF waste containers are to be cooled for 80 days, within the processing area. This calls for a cooling station with the capability of cooling eight NaF waste containers whose average age varies from about 7 days to 87 days. The following heat generation rates apply to the cooling station:

Identification Number of Containers	Average Heat Generation Rate (Btu/hr)
1 (about 7 days old)	616,800
2	380,900
3	261,200
4	210,400
5	163,600
6	145,100
7	119,700
8 (about 87 days old)	<u>104,000</u>
Total heat generation at cooling station	2,001,700 Btu/hr

Interim-Storage of Waste

As mentioned above, the waste containers are to be cooled for a minimum of 80 days by forced air convection prior to their transfer to the interim waste-storage facility. The average heat generation rate of one container at the end of this time (based on the slope of the fission-product decay heat for this period) will be 104,000 Btu/hr.

An average of 28 containers are to be sent to this waste-storage facility per year at one-month intervals — an average of 2.33 containers per month. When the cell is opened for transfer, the most recently filled container has 80-day-old material in it, whereas, the first filled is about 110 days old. On the average, the heat generating rate of the transferred containers is characteristic of one that is 95 days old whose rate is 87,500 Btu/hr. The average heat generation rate of containers sent to the interim storage facility is:

$$2.33 \times 87,500 = 203,900 \text{ Btu/hr} .$$

The heat generation rate of these containers decays with storage time, and the average rate in the interim facility for one year's accumulation is:

Residence Time of NaF Waste Container in Interim Facility (months)	Average Heat Generation Rate (Btu/hr)
1	203,900
2	150,200
3	102,000
4	83,700
5	69,800
6	62,200
7	53,700
8	42,900
9	37,600
10	33,300
11	29,000
12	26,800
Average heat generation rate of a one-year accumulation of containers	895,100 Btu/hr

For five years of waste accumulation, the total heat generation rate in the interim facility is:

Age of NaF Waste Tanks (years)	Average Heat Generation Rate (Btu/hr)
1	895,100
2	381,300
3	198,900
4	132,600
5	116,000
Heat generation rate from NaF waste in interim facility	1,723,900 Btu/hr

APPENDIX C. ESTIMATION OF DISTILLATION RATE IN VACUUM STILL

Here, calculations are given whereby the distillation rate of the vacuum still is estimated. The configuration of the still is shown in Fig. 9 of the text. The calculational method used is a modification of the procedure for calculating the rate in molecular distillation as given by Burrows.¹⁶ For our still, it is estimated that at 1000°C and at a pressure of 1 mm of Hg, the distillation rates for LiF and BeF₂ are, respectively, 3.32×10^{-3} and $2.02 \times 10^{-3} \text{ g sec}^{-1} \text{ cm}^{-2}$.

For molecular distillation, a still is designed so that its condensing surface is located quite close to its evaporating surface, thereby minimizing the transport distance for the vapor. If the separation distance is small enough and if the pressure is low enough, a molecule leaving the liquid surface has a very high probability of reaching the condenser without colliding with another molecule. This is the essence of molecular distillation. Our still cannot be described precisely as a molecular still because its operating pressure is too high; however, the pressure is low enough that conditions for molecular distillation are approached. For this reason, the calculational methods of molecular distillation, modified to apply to pressures slightly higher than those for true molecular distillation, are used here to estimate the distillation rate for our still.

The theoretical rate of distillation of a single substance for such conditions can be derived from the kinetic theory of gases, shown by Burrows to be:

$$5.83 \times 10^{-2} p_{\text{mm}} \sqrt{M/T} \text{ g sec}^{-1} \text{ cm}^{-2} \quad (\text{C-1})$$

where p_{mm} is the equilibrium vapor pressure in mm of Hg, M is the molecular weight in grams, and T is the absolute temperature in °K. The resulting distillation rate is expressed as $\text{g sec}^{-1} \text{ cm}^{-2}$.

The actual rate of distillation at low pressure will be less than the theoretical rate because there will be collisions in the vapor space. Burrows developed an expression for the factor by which the theoretical distillation rate should be multiplied to get the actual rate. Summarizing,

his treatment considers three events that can occur to the molecule in the vapor space:

1. Some molecules can reach the condenser without a collision; this fraction is e^{-K} , where K is a dimensionless factor depending upon the distance between evaporator and condenser, the equilibrium mean free path of the molecules, and the shape of the evaporating surface.
2. The fraction of molecules that collide is $(1 - e^{-K})$, and the fraction of these molecules that reach the condenser is $(1 - e^{-K})e^{-K}$, approximately.
3. From purely geometrical considerations, the probability of a molecule's striking the condenser after many collisions, which result in random motion, is a factor F, the ratio of condenser area to condenser area plus evaporator area. The fraction of these molecules that reach the condenser after many collisions is $F(1 - e^{-K})(1 - e^{-K})$.

The total fraction f of vaporized molecules that reach the condenser is given by the sum of the three fractions above:

$$\begin{aligned} f &= e^{-K} + (1 - e^{-K})e^{-K} + F(1 - e^{-K})(1 - e^{-K}), \\ &= F + (1 - F)(2e^{-K} - e^{2K}), \\ &= 1 - (1 - F)(1 - e^{-K})^2. \end{aligned} \quad (C-2)$$

When the theoretical rate is multiplied by f, defined to be the evaporation coefficient, the rate of distillation for a single component becomes:

$$5.83 \times 10^{-2} f p_{mm} \sqrt{M/T} \quad g \text{ sec}^{-1} \text{ cm}^{-2}. \quad (C-3)$$

The evaporation coefficient f is not readily calculated because of the difficulty in determining the proper value of K, which is defined to be:

$$K = \frac{d}{k\lambda_E}, \quad (C-4)$$

where

d = gap distance between evaporator and condenser,

λ_E = mean free path of molecules in equilibrium vapor,

k = suitable constant, which is 1 or larger, used to relate actual conditions to average conditions in the gap. Burrows¹⁶ reports experimental values of k in the range 3.5 to 30.

However, as K increases, the value of f approaches the value of F, a number that is known from the configuration of the still. For the still shown in Fig. 9 (see text), $F \approx 0.58$. An estimate of the size of K for this still can be obtained for the two values of k given above and from the fact that

$$\lambda_E = \frac{2.3 \times 10^{-2} T}{p_{mm} \sigma^2}, \text{ cm} \quad (C-5)$$

where

σ = molecular diameter, cm,

T = temperature, $^{\circ}$ K,

p_{mm} = equilibrium pressure, mm Hg.

Substituting into Eq. (C-4),

$$K = \frac{d p_{mm} \sigma^2}{2.3 \times 10^{-20} Tk}. \quad (C-6)$$

For LiF at $T = 1273^{\circ}$ K,

$\sigma \approx 3.26 \times 10^{-8}$ cm,

$p_{mm} = 1.0$ mm Hg,

and for this still, $d \approx 10$ cm. Using these values, one finds that

$K = 12.1$ for $k = 30$,

and

$K = 103.7$ for $k = 3.5$.

In either case, K is sufficiently large that Eq. (C-2) reduces to

$$f \approx F \approx 0.58.$$

In processing the fuel stream of the MSBR, distillation must treat a multicomponent mixture of LiF-BeF₂-UF₄-fission products. However, LiF and BeF₂ constitute more than 99 mole % of the mixture, as shown in Table C-1, making it possible to treat the mixture as a binary solution for purposes of calculating the distillation rate.

Burrows has shown that for a binary mixture, Eq. (C-3), which gives the distillation rate, becomes:

$$5.83 \times 10^{-2} f P_1 x_1 A \sqrt{M_1/T} \text{ g/sec}, \quad (C-7)$$

where

x_1 = mole fraction of component 1 in the liquid mixture,

A = area of evaporating surface, cm^2 ,

P_1 = equilibrium vapor pressure of component 1 at temperature T, mm Hg,

M_1 = gram-molecular weight of component 1.

A more rigorous representation of Eq. (C-7) would include the activity coefficient γ_1 of component 1 instead of the evaporation coefficient f to account for deviations from ideality. However, the activity coefficients are not known for these salt solutions, so for this study it will be assumed that Eq. (C-7) gives a reasonably valid estimate of the distillation rate.

Since the still operates at constant volume, a material balance requires that BeF_2 and LiF distill at the same rate at which they enter the still. The vapor composition is therefore about 69-31 mole % LiF- BeF_2 , the same as it is in the stream entering the still. If the total pressure in the vapor space of the still is kept at 1 mm Hg, the partial pressures of LiF and BeF_2 are 0.69 and 0.31 mm Hg, respectively, assuming that there

Table C-1. Reactor Data and Approximate Composition of Fuel Stream at Equilibrium

Reactor Data

Fuel volume	=	671 ft ³
Cycle time	=	58 days
Power	=	2160 Mw (thermal)

Approximate Composition (mole fraction)

^{233}U	0.0029
^{235}U	Negligible
Other U	0.0002
LiF	0.6840
BeF_2	0.3118
Fission products	0.0008 ^a

^aGaseous fission products purged in the reactor circulating loop, and noble fission products that are removed on a very short cycle by attaching themselves to the INOR-8 walls, do not contribute to this value.

are no other volatile compounds in the still. This assumption is valid because the quantities of volatile fission products are quite small. If we assume that Raoult's law applies, the partial pressure of BeF_2 is:

$$P_{\text{BeF}_2} = P_{\text{BeF}_2}^{\circ} x_{\text{BeF}_2}$$

The vapor pressure of BeF_2 at 1000°C is about 74 mm Hg; therefore the mole fractions in the still liquid are approximately:

$$x_{\text{BeF}_2} = 0.31/74 = 0.0042,$$

and

$$x_{\text{LiF}} = 1 - x_{\text{BeF}_2} = 0.9958.$$

By Eq. (C-7), the rates at which LiF and BeF_2 evaporate from a surface having area A can be determined:

$$w_{\text{LiF}} = 5.83 \times 10^{-2} fA[xP\sqrt{M/T}]_{\text{LiF}} \text{ g/sec}, \quad (\text{C-8})$$

and

$$w_{\text{BeF}_2} = 5.83 \times 10^{-2} fA[xP\sqrt{M/T}]_{\text{BeF}_2} \text{ g/sec}. \quad (\text{C-9})$$

The vapor pressure of LiF at 1000°C is 0.61 mm Hg. When this value and the values of the other quantities are inserted in Eqs. (C-8) and (C-9), the specific evaporation rates are found to be:

$$3.32 \times 10^{-3} \text{ g sec}^{-1} \text{ cm}^{-2} \text{ for LiF},$$

and

$$2.02 \times 10^{-3} \text{ g sec}^{-1} \text{ cm}^{-2} \text{ for } \text{BeF}_2.$$

Adding these last two equations and solving for the area of the evaporating surface:

$$A = \frac{w_{\text{LiF}} + w_{\text{BeF}_2}}{5.83 \times 10^{-2} f[(xP\sqrt{M/T})_{\text{LiF}} + (xP\sqrt{M/T})_{\text{BeF}_2}]}, \quad (\text{C-10})$$

$$A = \frac{(436,600 + 359,000)}{(24)(3600)}, \frac{\text{g}}{\text{sec}}$$

$$(5.83 \times 10^{-2})(0.58)[(0.9958)(0.69)\sqrt{26/1273} + (0.0042)(74)\sqrt{47/1273}], \frac{\text{g}}{\text{sec cm}^2}$$

$$= 1720 \text{ cm}^2,$$

$$\approx 1.85 \text{ ft}^2.$$

Now, the area of the pool surface in the still (see Fig. 9 in text) is 4.9 ft². However, it is not safe to say that the still is overdesigned by a factor of about 2.5. Better data on vapor pressures and particularly activity coefficients might make a considerable difference in the calculated distillation rate. It is strongly suggested that the still considered for this study be viewed only as an approximate design that will probably change as more is learned about the distillation process.

APPENDIX D. FISSION-PRODUCT ACCUMULATION AND HEAT GENERATION RATE IN LITHIUM FLUORIDE POOL IN VACUUM STILL

Nonvolatile fission products accumulate in the vacuum still as the LiF-BeF₂ fuel carrier and volatile fission products are distilled away. The still is initially charged with 4 ft³ of makeup ⁷LiF, evacuated to a pressure of 1 mm Hg or less and adjusted to a temperature of about 1000°C. Fluorinated fuel salt, containing fission products, is allowed to flow continuously into the pool of LiF, and the still is operated so that the rate of distillation is exactly equal to the feed rate. There is no bottom discharge, and the volume remains constant. Kelly's work²⁵ showed that the initial LiF and BeF₂ distillate is decontaminated by a factor of 100 to 1000 from rare earths. Operation is continued at the above pressure and temperature until one of two phenomena forces termination: (1) Either the solubility of fission products in the 4 ft³ of LiF is exceeded and troublesome precipitation occurs, or (2) the accumulated heat generation rate from fission-product decay begins to tax the capacity of the cooling system. When the distillation is terminated, the LiF-fission-product residue in the still is drained to a waste receiver and eventually to permanent storage. The operating cycle is then repeated.

The aim of the following calculations is to determine the operating cycle for the still and the limiting conditions for the design shown in Fig. 9. Since one design criterion is to process the fuel stream with minimum out-of-reactor holdup, and since the solubility of the rare earths in LiF at 1000°C is about 50 mole %, it is apparent that condition (2) above will be controlling. Decay-heat removal will be a serious problem long before solubility limits are approached. It has been determined that the 4-ft³ still can operate continuously for about 67.4 days at a distillation rate of 15 ft³/day, processing fuel that has the heat generation characteristics shown in Figs. 2a and b of the text. The significance of this number is that it represents the rate of ⁷Li discard to waste - 116 kg every 67.4 days. The heat generation rate at this time is 31×10^6 Btu/hr, the maximum that can be removed by the cooling system.

The calculations below are believed to give a conservative estimate of the still performance. The basic data on fission-product heat generation

versus decay time represent gross values and hence do not exclude the contributions of those fission products removed by the reactor gas sparge, by deposition on metal surfaces in the reactor system, or by the fluorination step in chemical processing. Furthermore, the 15-ft³/day design rate is probably excessive for a 1000-Mw (electrical) MSBR; the economic optimum rate is perhaps nearer 12 ft³/day. Also, no credit was taken for periodic interruptions in processing due to the reactor's operating at less than 100% plant factor. All these factors tended to shorten the still operating cycle and increase the discard cost.

The calculations are arranged as follows: First, an analytical expression is derived for the heat generation rate of fission products in the still as a function of elapsed time since discharge from the reactor. Second, this equation is then used to evaluate the still design.

Analytical Expression for Heat Generation Rate

When irradiated fuel salt is discharged from the fluorinator, it has been out of the reactor about 38 hr. A cooling tank downstream from the fluorinator adds another 24 hr of holdup, so that the salt is about 62 hr old when it reaches the still. The heat generation characteristics of the salt during this period were calculated by assuming an infinitely mixed system; these calculations are described in Appendix A, and the results are shown graphically in Fig. A-1. Continuous operation is assumed throughout the system, and, by the time the fuel reaches the still, its heat generation (Fig. A-1) is about $4.45 \times 10^4 \text{ Btu hr}^{-1} \text{ ft}^{-3}$.

The still is a sink for fission products that exhibit decay behavior like that shown in Fig. 2a and b. Initially at $t = 62 \text{ hr} = 2.58 \text{ days}$, the 4 ft³ of LiF in the still is fresh material having a zero heat generation rate, but this condition changes rapidly when the still is put on-stream. Flow into the still is continuous, and the rate of heat generation will rise until flow is stopped. The magnitude of this rate at any future time T is therefore an integrated quantity over the accumulation period: $t_0 = 2.58 \text{ days to } T$. To describe the behavior of the still during this period, define the quantities

- q_o = specific heat generation rate of fuel entering still,
 $= 4.45 \times 10^4 \text{ Btu hr}^{-1} \text{ ft}^{-3}$,
- $q(t)$ = specific heat generation rate at time t , $\text{Btu hr}^{-1} \text{ ft}^{-3}$,
- F = flow rate of fuel into still,
 $= 15 \text{ ft}^3/\text{day}$,
- $v(t)$ = fuel volume processed at time t ,
- t = time that still has been operating, days,
- v_s = volume fuel pool in still,
 $= 4 \text{ ft}^3$.

The heat generation in the still pool at any time t is $v_s q_s(t)$, expressed as Btu/hr, which is the difference between what the rate would have been if there had been no decay and the amount the rate has decreased because of decay. That is,

$$v_s q_s(t) = \int_v q_o dv(t) - \int_v q(t) dv(t). \quad (\text{D-1})$$

Noting that

$$dv = F dt, \quad (\text{D-2})$$

there results

$$v_s q_s(t) = F q_o \int_t^\infty dt - F \int_t^\infty q(t) dt. \quad (\text{D-3})$$

The limits of integration extend from $t_0 = 2.58$ days to $t = T$ days, where T denotes the time after discharge from the reactor at which the heat generation rate is desired.

To treat Eq. (D-3) analytically, the function $q(t)$ is obtained from Figs. 2a and b by representing the curve on this graph by four straight-line segments in the range 2 to 400 days. The procedure is diagrammed in Fig. D-1. The general form of the equations is:

$$q(t) = k t^n.$$

The slope n of each segment is determined from values of $q(t)$ read directly from Figs. 2a and b. The initial conditions of the fuel entering the still, namely, $t = 2.58$ days, $q = 4.45 \times 10^4 \text{ Btu hr}^{-1} \text{ ft}^{-3}$, were introduced into the equation for the first segment to determine the constant k_1 . The constant k_2 was determined similarly by using the end condition of the

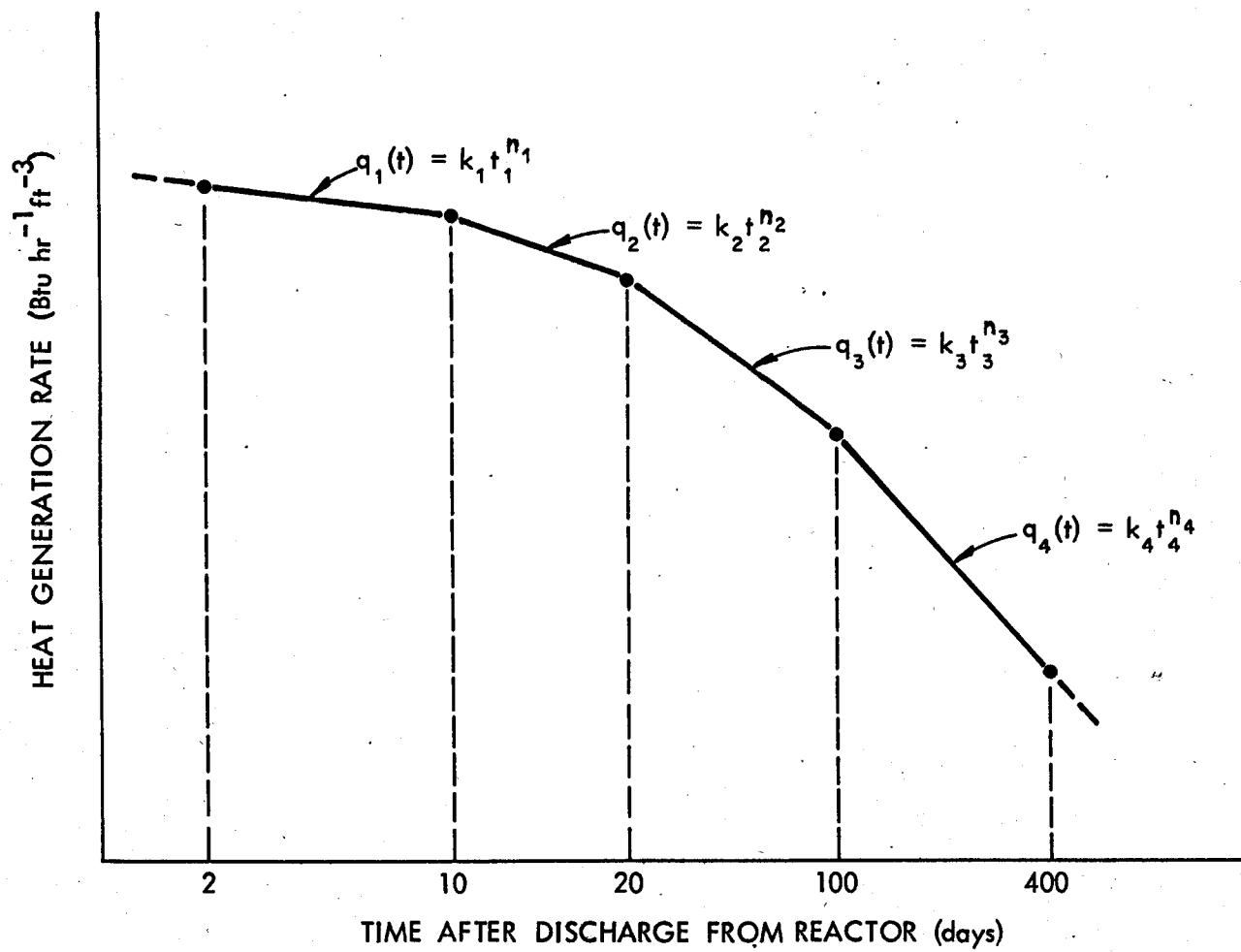


Fig. D-1. Schematic Logarithmic Curve Showing Approximations to Fission-Product Decay. In the mathematical model for computing the heat generation rate, it is convenient to divide the time scale into four parts.

first segment at $t = 10$ days. This stepwise procedure was followed for segments two and three to evaluate k_3 and k_4 . The four equations so determined are:

$$q_1 = 6.69 \times 10^4 t_1^{-0.43} \quad 2 \leq t_1 \leq 10 \text{ days}, \quad (\text{D-4})$$

$$q_2 = 9.686 \times 10^4 t_2^{-0.59} \quad 10 \leq t_2 \leq 20 \text{ days}, \quad (\text{D-5})$$

$$q_3 = 21.2 \times 10^4 t_3^{-0.854} \quad 20 \leq t_3 \leq 100 \text{ days}, \quad (\text{D-6})$$

$$q_4 = 194.8 \times 10^4 t_4^{-1.335} \quad 100 \leq t_4 \leq 400 \text{ days}. \quad (\text{D-7})$$

In these equations, q is in $\text{Btu hr}^{-1} \text{ ft}^{-3}$, and t is in days.

The function $q(t)$ in Eq. (D-3) is replaced by the four separate functions of Eqs. (D-4 to (D-7), and the integration is carried out to give:

$$\begin{aligned} V_s q_s(t) = & 15 \times 10^4 [4.45(T - 2.58) - 11.74(t_1^{0.57} - 2.58^{0.57}) \\ & - 23.62(t_2^{0.41} - 10^{0.41}) - 145.2(t_3^{0.146} - 20^{0.146}) \\ & + 581.5(t_4^{-0.335} - 100^{-0.335})]. \end{aligned} \quad (\text{D-8})$$

The units of T , t_1 , t_2 , t_3 , and t_4 are days; the units of $V_s q_s$ are Btu's/hr. The restrictions on the several t 's are those specified for Eqs. (D-4) to (D-7). In solving Eq. (D-8) for $V_s q_s(T)$, a value is chosen for T , and this value is assigned to either t_1 , t_2 , t_3 , or t_4 in the following way: For

$$2.58 < T \leq 10 \text{ days, let } t_1 = T;$$

$$10 < T \leq 20 \text{ days, let } t_2 = T;$$

$$20 < T \leq 100 \text{ days, let } t_3 = T;$$

$$100 < T \leq 400 \text{ days, let } t_4 = T.$$

All $t_i < T$ assume their maximum values; all $t_i > T$, of course, are not considered. Figure D-2 is a plot of Eq. (D-8); it gives the integrated heat generation rate for times between 2.58 and 400 days after discharge from the reactor. Note that the accumulation time for fission products in the still is $(T - 2.58)$ days.

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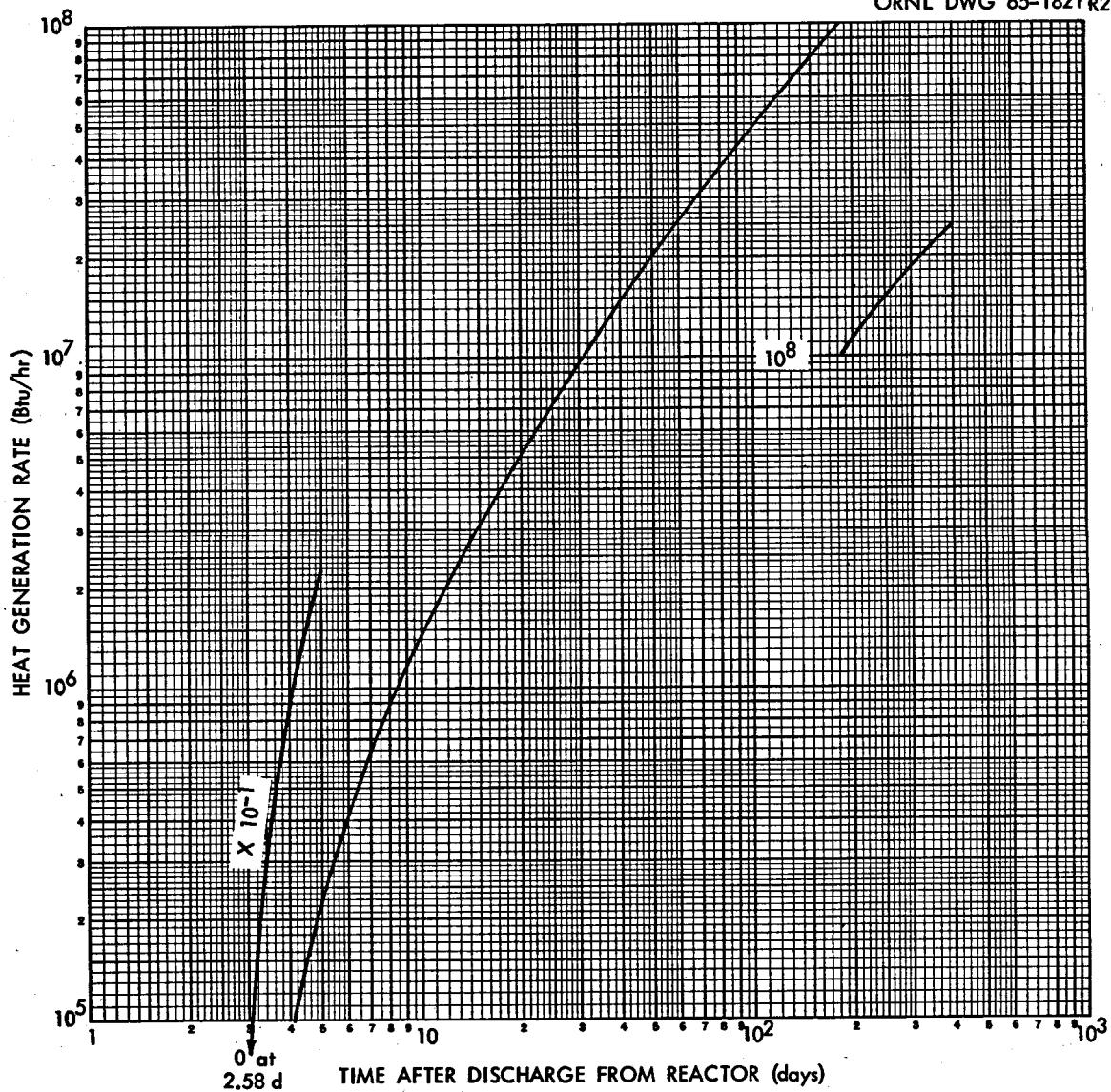


Fig. D-2. Heat Generation Rate in the LiF Pool Resulting from Fission-Product Accumulation in the Still. The still is charged with 4 ft³ of fresh, molten LiF at the beginning of the distillation cycle. Barren fuel-carrier (2.58 days old) flows into this LiF at a rate of 15 ft³/day, and LiF-BeF₂ distills at the same rate, keeping the volume constant. Accumulating fission products cause a rapid increase in the heat generation rate.

Evaluation of Vacuum-Still Design

Two principal conditions had to be satisfied in the still design:

(1) the evaporating surface had to be sufficient for obtaining the required 15-ft³/day distillation rate, and (2) maximum heat transfer surface had to be provided to minimize the frequency with which the still is drained to the waste tank. The choice of dimensions was somewhat arbitrary; a tube-sheet diameter of 2.5 ft was chosen because it gave an evaporating surface that was about 2.5 times the calculated area, and a closely spaced arrangement of 2.5-ft-long by 1/2-in.-diam tubes was used for high cooling-surface-to-volume ratio. The primary unknown operating condition is the length of time that fission products can be accumulated before the integrated heat generation taxes the capacity of the cooling system; an estimate of this time is determined in the following calculations. The schematic diagram in Fig. D-3 depicts the still operation and summarizes calculated performance and physical data.

The calculations which are described below indicate that the still can accumulate fission products for about 67.4 days in a 4-ft³ LiF pool when the still is operated continuously at a feed rate of 15 ft³/day. The discard rate for LiF is therefore a very small fraction of the processing rate, being about 0.4%. As mentioned above, the heat generation rate predicted by Eq. (D-8) is probably excessive because gross instead of net fission-product data were used; therefore this accumulation time can be treated as a lower limit.

Physical Data for Still

Applicable data are given on Fig. D-3.

Heat Transfer Characteristics

The 1000°C temperature of the LiF pool in the still was chosen to achieve adequate distillation rate and fission-product solubility. The pool transfers heat to the tube walls by natural convection, and this heat is picked up on the outside of the tubes by NaK (22.3-77.7 mole % Na-K eutectic) under forced convection. The NaK coolant enters the still at

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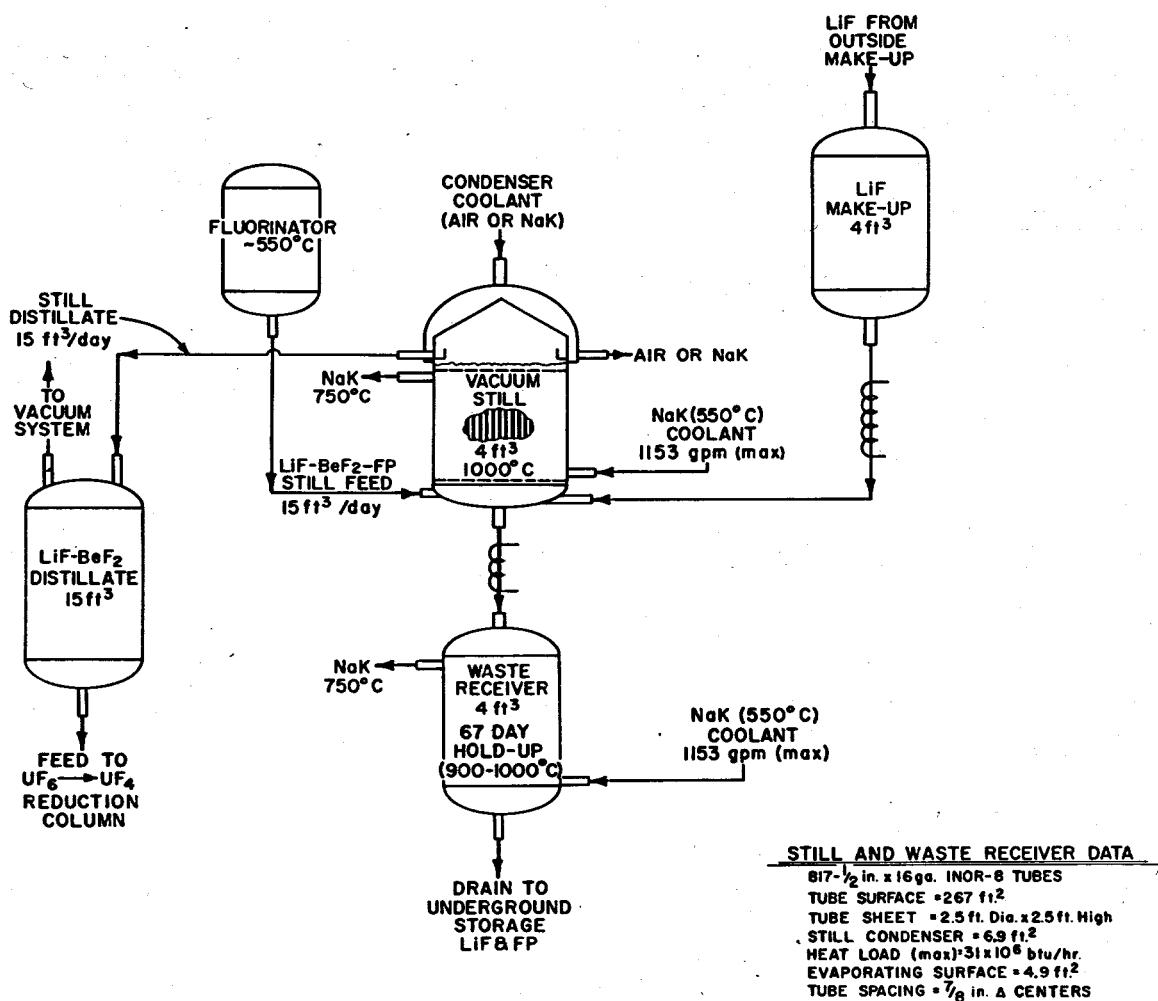


Fig. D-3. Schematic Diagram of Vacuum Still Operation.

550°C and exits at 750°C, giving a logarithmic mean temperature difference of 340°C (612°F). Heat transfer characteristics of LiF were calculated by using the Nusselt equation for natural convection, in conjunction with the physical-property data given in Appendix F. A Nusselt-type equation for flow normal to banks of tubes was used for calculating the heat transfer properties of the coolant. Since the total heat dissipation requirement of the still was unknown (because fission-product accumulation time was not known), the coolant heat transfer coefficient was expressed as a function of the heat generation rate. It was determined that the still could dissipate about 31×10^6 Btu/hr for the NaK flow conditions shown in Fig. D-3. The overall heat transfer coefficient for maximum heat flux is about $190 \text{ Btu hr}^{-1} \text{ ft}^{-2} \text{ }^{\circ}\text{F}^{-1}$. From Fig. D-2, the time after discharge from the reactor corresponding to the above integrated heat generation is 70 days. The fission-product accumulation time is 2.58 days less or 67.4 days because of the time lag before fuel reaches the distillation step.

LiF, BeF₂, UF₄, and Fission-Product Discard Rates. — At the end of the 67.4-day cycle, the still contents are primarily LiF and fission products. The equilibrium BeF₂ concentration was estimated above to be only 0.4 mole % because of its relatively high vapor pressure at 1000°C. If it is assumed that the fluorination step is 99.7% efficient for uranium removal, then in 67.4 days about 3.64 kg of U will have entered the still. Uranium tetrafluoride has a vapor pressure at 1000°C about 2.3 times that of LiF, so a portion of the UF₄ will be recovered in the distillate. The amount recovered cannot be calculated until more is known about the vapor-liquid equilibria of multicomponent molten salt mixtures. Fission-product accumulation during this period is approximately 190.1 kg; most of the RbF and CsF of this inventory will distill because of relatively high vapor pressures.

An estimate of the inventory relegated to waste every 67.4 days is given in Table D-1. A reactor plant-factor of 100% was assumed in the calculation. These values are based on ideal-solution behavior, particularly with regard to LiF and BeF₂; this is almost surely not the case for this mixture. As more is learned about the activities of the components, it can be expected that the compositions of Table D-1 will be different from those shown.

Table D-1. Estimated Composition of the Vacuum Still
After 67.4 Days' Operation at 15 ft³/Day

Still volume = 4 ft³

	Weight (kg)	Mole Fraction
^{233}U	<3.33	<0.002
^{235}U	<0.05	-
Other U	<0.26	-
LiF	195.9	0.850
BeF_2	2.1	0.005
Fission products ^a	190.1	0.143

^a Molecular weight = 150, assumed.

Heat Removed by Still Condenser. - The load on the still condenser is relatively small, consisting only of the latent heat in 15 ft³/day of a 69-31 mole % LiF- BeF_2 mixture plus some radiation from the pool surface. The latent heat of vaporization of LiF was estimated from vapor pressure data to be 44,000 cal/g-mole; an experimental value¹¹ of 50,100 cal/g-mole was used for BeF_2 . Radiative heat transfer amounted to about 62,700 Btu/hr, giving a total condenser duty of 198,300 Btu/hr.

For smooth, nonflashing distillation, the condensate is the LiF- BeF_2 eutectic having a melting point about 500°C, and a condenser temperature slightly higher than this is satisfactory. If there are deviations from ideal operation so that a higher-melting composition distills, the condenser temperature could be adjusted to temperatures slightly above the melting points of the pure components, 803°C for BeF_2 and 845°C for LiF.

APPENDIX E. DESIGN CALCULATIONS FOR WASTE-STORAGE SYSTEM

Separate storage is provided for fuel and fertile stream wastes as fluorides in underground tanks designed for bulk accumulation over a 30-year period. Waste management in the post-30-year period was not considered in this study. However, as mentioned in the text (see Table 2), it will probably be desirable to reprocess the fertile-stream waste, which is only mildly radioactive, at some future time for recovery of thorium, lithium, and uranium values. About 116 kg of ^{233}U will be present. Fission products, separated in this recovery, could be stored longer, depending on the activity.

In the fuel-stream waste, the only significant value, other than the possible future value of individual fission products, is ^7Li . However, as explained below, it is necessary to add a mixture of NaF-KF to this waste to facilitate heat transfer. Since these compounds are chemically similar to LiF, recovery of the lithium is difficult. In any event, at the end of the 30-year period, the desirability of recovery would have to be analyzed in light of the prevailing costs.

Two basic problems must be solved in designing the waste storage system: (1) the integrated rate of heat generation by fission-product decay must be determined, and (2), using the results of (1), the most economic design for the prevailing conditions must be found. The heat generation rate is computed from the fission-product decay behavior exhibited in Figs. 2a and b (see text), and the results are shown in Figs. E-1 and E-2. A previous study by Carter and Ruch¹⁴ examined a similar waste-storage problem, and, in accord with their recommendation for economic waste management, bulk storage in large, heat-exchanger tanks is adopted for our wastes. Conceptual designs of the waste facilities are shown on Dwgs. 58080 D and 58081 C in Appendix F.

Over the 30-year period, 784 ft^3 of fuel-stream waste are collected. This volume includes 520 ft^3 of LiF-fission-product mixture drained from the processing cell, plus 284 ft^3 of NaF-KF diluent. The storage tank is 16 ft in diameter and 6.33 ft high. The corresponding volume of fertile-stream waste is 1783 ft^3 ; this is stored in a tank 13.5 ft in diameter by 13.5 ft high.

Most of the following calculations are concerned with the design of the waste tank for fuel-stream effluent because the most difficult design problems occur for this waste. After the decay characteristics of the fission products and integrated heat generation rate are determined, the calculations examine, in the following order, the basic features of tank design, maximum allowable heat generation rate, volume of diluent, waste-tank design, and the underground storage facility. Once these calculations are made, computations for the fertile-stream waste are almost incidental.

Fuel-Stream Waste System

Decay Characteristics of the Fission Products

The initial problem in designing the waste tank is to determine the decay in terms of the time-related characteristics of the salt being added to the tank. That is to say, the still bottoms, which represent an accumulation of fission products having ages in the range 2.58 to 70 days, generate heat at a rate characteristic of fission products having an "average" age somewhere between these values. The time-related behavior of the integrated, specific heat generation rate is shown in Fig. 11 of the text for the case of no inert salt dilution. The graph covers a 5-year collection period of 4-ft³ batches every 67.4 days. The initial point on the graph begins at about 140 days; this is the accumulated time in the processing plant with a reference time of zero taken as the day the fuel is discharged from the reactor. An in-cell delay period of 67.4 days after removal from the still is included in this time to permit some initial cooling before draining to the underground tank.

The decay curves of Figs. 2a and b can be used to determine the decay behavior when this "average" age has been found. The volume of fuel salt from which fission products have accumulated is:

$$15 \text{ ft}^3/\text{day} \times 67.4 \text{ days} = 1011 \text{ ft}^3.$$

It was determined in Appendix D that the heat generation rate at age equal to 70 days out of the reactor is 31×10^6 Btu/hr. Therefore on the basis of unit volume of core salt the average heat generation rate is:

$$q_{\text{avg}} = \frac{31 \times 10^6}{1011} = 3.066 \times 10^4 \text{ Btu hr}^{-1} \text{ ft}^{-3}.$$

Define the average heat generation,

$$q_{\text{avg}} = \frac{\int q dv}{v} , \quad (\text{E-1})$$

where v is the volume. The processing rate F , in ft^3/hr , is steady, so in the time interval dt ,

$$dv = F dt . \quad (\text{E-2})$$

It was shown in Appendix D that the instantaneous heat generation rate is represented by an equation of the form:

$$q = kt^{-n} , \text{ Btu hr}^{-1} \text{ ft}^{-3} \quad (\text{E-3})$$

Combining Eqs. (E-2) and (E-3) in (E-1):

$$q_{\text{avg}} = \frac{k \int_{t_0}^{t_{\text{avg}}} t^{-n} dt}{\int_{t_0}^{t_{\text{avg}}} dt} ,$$

$$q_{\text{avg}} = \frac{k \left[t_{\text{avg}}^{(1-n)} - t_0^{(1-n)} \right]}{(1-n)(t_{\text{avg}} - t_0)} , \quad (\text{E-4})$$

where k and n are characteristic constants for the decay curve; t_0 ($= 2.58$ days) is the age of the fission products at the beginning of distillation. The desired average age is t_{avg} , the time corresponding to q_{avg} .

It will be recalled from the discussion of Appendix D (in particular Fig. D-1) that the q -versus- t curve is best represented by four segments over the range of interest. The four equations are:

$$q_1 = 6.69 \times 10^4 t_1^{-0.43} \quad 2 \leq t_1 \leq 10 \text{ days}, \quad (\text{E-5})$$

$$q_2 = 9.686 \times 10^4 t_2^{-0.59} \quad 10 \leq t_2 \leq 20 \text{ days}, \quad (\text{E-6})$$

$$q_3 = 21.2 \times 10^4 t_3^{-0.854} \quad 20 \leq t_3 \leq 100 \text{ days}, \quad (\text{E-7})$$

$$q_4 = 194.8 \times 10^4 t_4^{-1.335} \quad 100 \leq t_4 \leq 400 \text{ days}. \quad (\text{E-8})$$

If t_{avg} lies in the range 2 to 10 days, then its value can be found directly from Eq. (E-4) by utilizing $k = 6.69 \times 10^4$ and $n = 0.43$ from Eq. (E-5).

However, if t_{avg} is in the range 10 to 20 days, Eq. (E-4) contains a second term in the numerator involving the constants $k = 9.686 \times 10^4$ and $n = 0.59$ of Eq. (E-6). Similarly for t_{avg} in the range 20 to 100 days, the numerator of Eq. (E-4) contains a third term involving the constants of Eq. (E-7).

The solution is easily determined by trial, and for this case the proper form of Eq. (E-4) contains the constants k and n from Eqs. (E-5) and (E-6):

$$q_{avg} = \frac{\frac{k_1}{1-n_1} (t_{10}^{1-n_1} - t_o^{1-n_1}) + \frac{k_2}{1-n_2} (t_{avg}^{1-n_2} - t_{10}^{1-n_2})}{t_{avg} - t_o}, \quad (E-9)$$

$$3.066 \times 10^4 = \frac{\frac{6.69 \times 10^4}{0.57} (10^{0.57} - 2.58^{0.57}) + \frac{9.686 \times 10^4}{0.41} (t_{avg}^{0.41} - 10^{0.41})}{t_{avg} - 2.58}.$$

The solution to this equation is: $t_{avg} = 11.6$ days. An average age near the lower end of the time scale would be expected because of the larger contribution by the "younger" fission products to heat generation. The quantity $t_{10} = 10$ days used in Eq. (E-9) denotes the upper limit of Eq. (E-5) and the lower limit of Eq. (E-6).

Integrated Heat Generation in Waste Tank

When the vacuum still is taken off stream and drained, its 4-ft³ volume is generating heat at 31×10^6 Btu/hr, but this rate is decreasing in a manner characteristic of 11.6-day-old fission products, as shown on Fig. 2b. At this point, the change in rate is rather rapid, indicating that a short delay time in the cell before draining to underground storage will appreciably alleviate design requirements for the large tank.

Installation of a second 4-ft³ vessel quite similar to the vacuum still is a convenient way of providing a 67.4-day holdup, during this time the heat generation rate decreases by a factor of about 4.5.

A decay curve was calculated for a typical batch of still residue by assuming that the curve was parallel to the decay curve of Fig. 2b for MSBR fission products after 11.6 days. The 67.4-day in-cell delay makes the accumulated time from reactor discharge to underground storage equal to

137.4 days; at this time the heat generation rate of the 4-ft³ batch is 6.8×10^6 Btu/hr. The curves of Fig. E-1 are plotted to show the thermal history of the underground waste tank over a 5-year collection period during which 27 batches are added; also the decay in the post-5-year collection period is shown. The curve with positive slope represents the buildup of heat generation rate from all batches accumulated up to the indicated time. This curve is actually a stepped curve, but for convenience it has been drawn smooth through the maximum point of each step. The waste tank is designed for 30 years' collection, and the accumulated heat generation at this time can be obtained with little error by extrapolating the positive-sloped curve of Fig. E-1.

Figure E-2 has been included to show the integrated heat generation rate in the underground waste tank when the still residue is drained into the tank immediately upon completion of the distillation cycle.

Figures E-1 and E-2 represent an upper limit for fission-product heat generation. As pointed out earlier in this report, the values are for gross fission-product decay, which includes nuclides that have been removed prior to vacuum distillation. Also the curves were calculated for a reactor plant factor of 100%.

Basic Features of Tank Design

A previous study¹⁴ showed that bulk storage in large tanks is the most economical management for fluoride wastes. A 30-year period was chosen because it coincided with the amortization period of the reactor plant, and a single tank is sufficient because the overall waste volume is small. For economy and reliability, cooling by forced air draft was adopted, and an upper limit of 60 ft/sec was assigned to the velocity. Waste temperature in the tank is not to exceed 750°C.

Maximum Allowable Heat Generation Rate

Under the above ground rules it is apparent that there is an upper limit that can be tolerated for the volumetric heat production rate, and that the waste will have to be diluted to wet sufficient cooling surface. In 30 years at 80% reactor plant factor there will be 520 ft³ of radioactive

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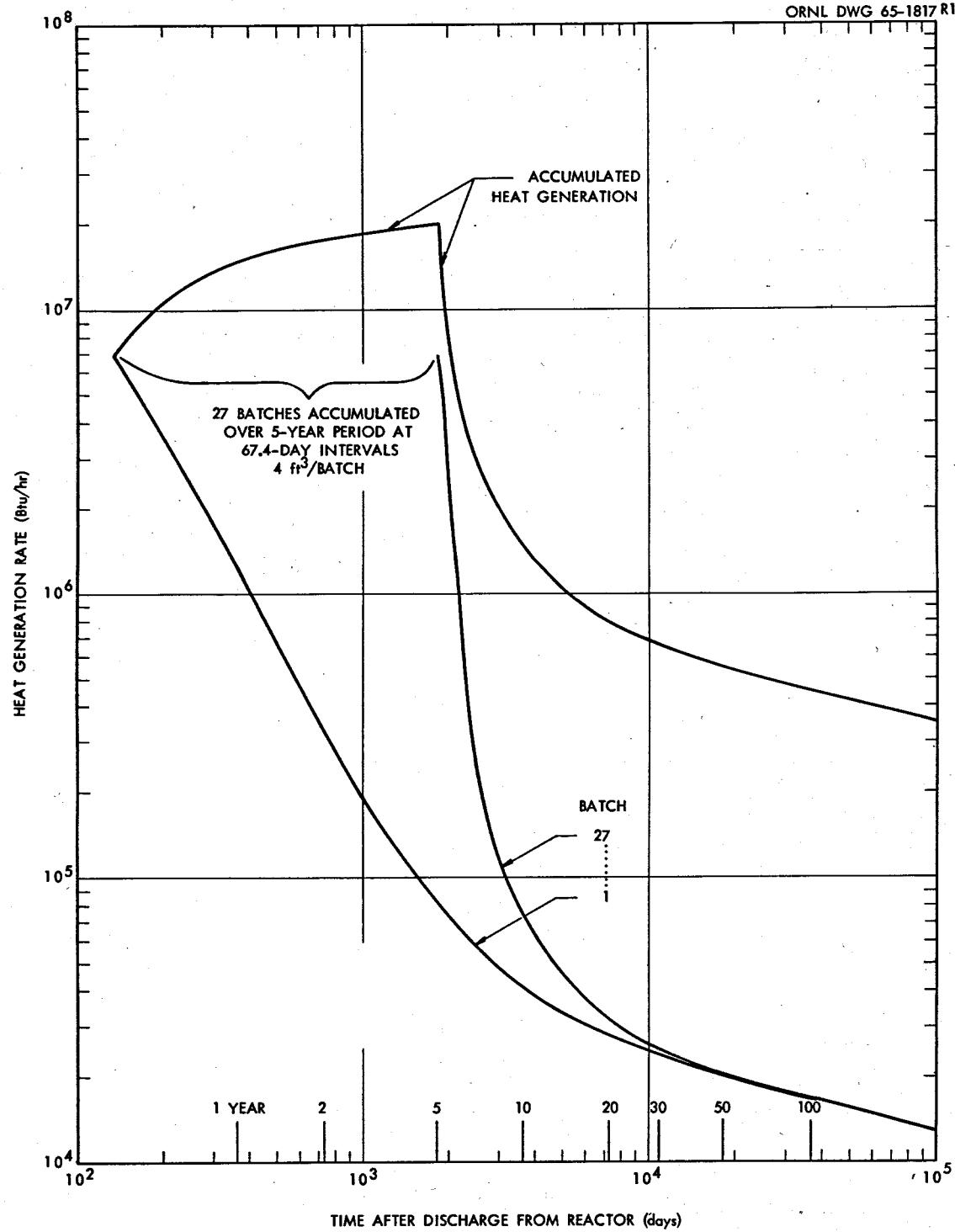


Fig. E-1. Decay Curves for Fuel-Stream Fission Products Cooled 67.4 Days Before Draining to Waste Tank. The total heat generation rate in the underground waste tank is significantly decreased by the 67.4-day holdup in the processing cell. Compare with Fig. E-2. Waste is from an MSBR operating on a 58-day cycle at 2160 Mw (thermal).

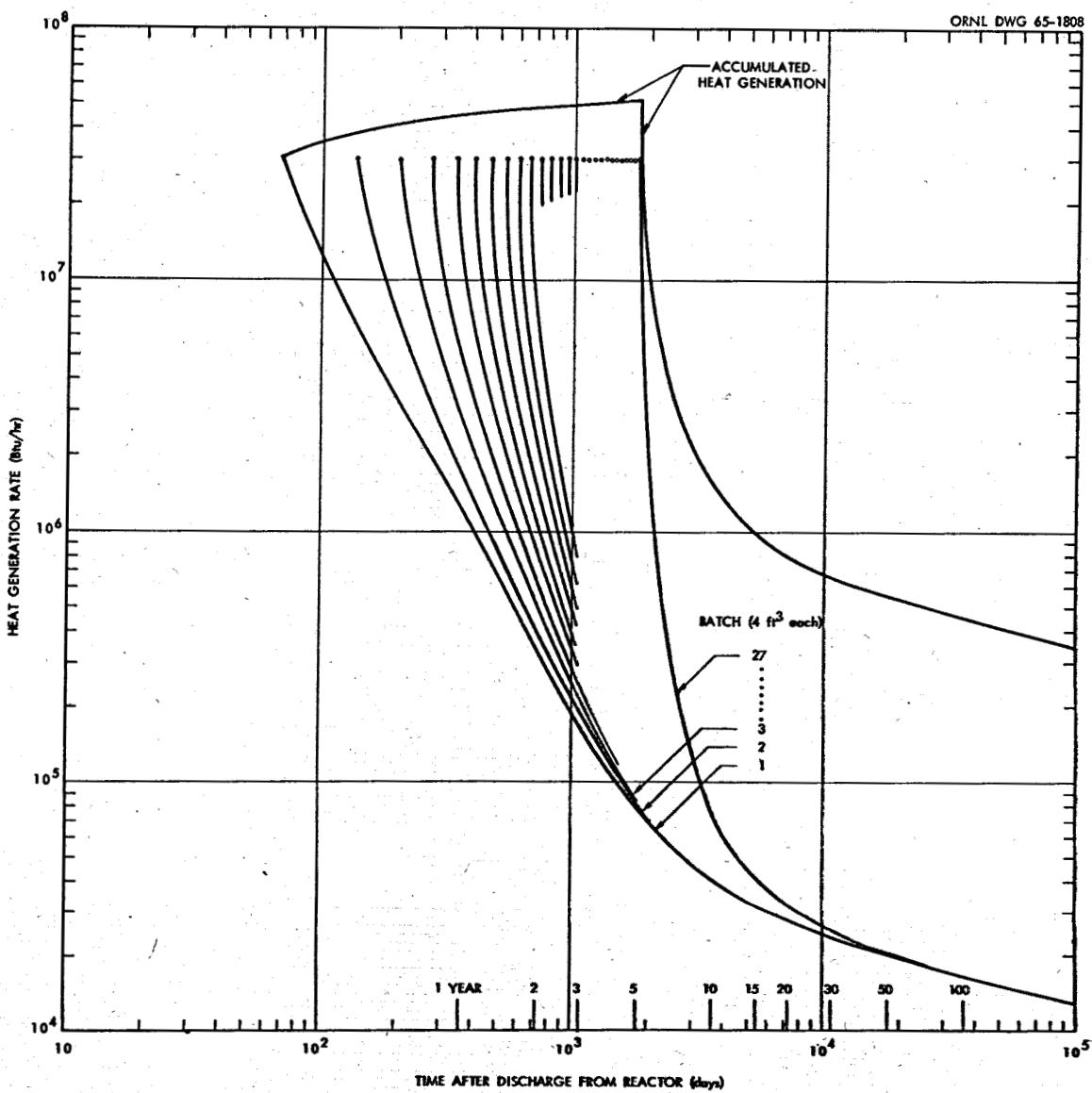


Fig. E-2. Decay Curves for Fuel-Stream Fission Products Sent to Waste Tank Without Prior Cooling. When the still bottoms are drained directly to the underground waste tank, the total heat generation rate is about 2.8 times that shown on Fig. E-1, in which a 67.4-day cooling is allowed. The cycle time and power of the reactor is the same as for Fig. E-1.

waste salt accumulated whose heat generation rate is about 1.76×10^7 Btu/hr, obtained by extrapolating the upper curve of Fig. E-1. Allowing the coolant to undergo a temperature rise of 125°C , the air requirement is:

$$1.76 \times 10^7 \frac{\text{Btu}}{\text{hr}} \times \frac{1 \text{ lb } ^{\circ}\text{F}}{0.25 \text{ Btu}} \times \frac{1}{(1.8)(125) ^{\circ}\text{F}} = 3.13 \times 10^5 \text{ lb/hr}.$$

This is equivalent to 85,190 ft³/min at an average temperature of 87°C .

The required cross section for air flow is:

$$\frac{85,190 \text{ ft}^3}{60 \text{ sec}} \times \frac{\text{sec}}{60 \text{ ft}} = 23.7 \text{ ft}^2.$$

This area can be obtained with 894 tubes 2.5 in. in diameter, with a 9-gage wall; in the final design, Drawing 58080 D, Appendix F, 937 tubes were used. These tubes, arranged in a U-tube configuration, can be accommodated in a 16-ft-diam tank.

Waste salt is stored on the shell side in the tank. The available volume per foot of tank is:

$$\begin{aligned} &= \frac{\pi D_{\text{tank}}^2}{4} \left[1 - \frac{(\text{No. tubes})d_{\text{tube}}^2}{D_{\text{tank}}^2} \right], \\ &= \frac{\pi}{4} (16)^2 \left[1 - \frac{(2 \times 937)(2.5)^2}{(16 \times 12)^2} \right] = 137.2 \text{ ft}^3/\text{ft}. \end{aligned}$$

The heat transfer surface per foot of tank height is:

$$= (2)(937)(0.577), \frac{\text{ft}^2}{\text{ft}} = 1081.3, \text{ ft}^2/\text{ft}.$$

Therefore, the surface available to each cubic foot of waste is $7.88 \text{ ft}^2/\text{ft}^3$.

The mean temperature difference between coolant and salt is 651°C , and the overall heat transfer coefficient is estimated to be $6.2 \text{ Btu hr}^{-1} \text{ ft}^{-2} ^{\circ}\text{F}^{-1}$. The maximum allowable heat generation is then:

$$\frac{6.2 \text{ Btu}}{\text{hr ft } ^{\circ}\text{F}^2} \times \frac{7.88 \text{ ft}^2}{\text{ft}^3} \times (651)(1.8) ^{\circ}\text{F} = 5.75 \times 10^4 \text{ Btu hr}^{-1} \text{ ft}^{-3}.$$

At all times during waste accumulation, there must be sufficient volume present so that the integrated heat generation (Btu/hr) divided by the total volume (ft³) does not exceed this figure.

Volume of Diluent

The total volume of fluid in the waste tank at any time is the summation of individual batch volumes from the process plus the required diluent. Expressed mathematically,

$$V(t) = \sum_{n=1}^N (4 + v_{dn}) ,$$

where v_{dn} is the required diluent volume for the nth batch, and the summation is carried out over N batches. The above paragraph shows that the total volume at any time is the quotient of:

$$\frac{\text{integrated heat generation rate (Btu/hr)}}{\text{allowable volumetric heat generation (Btu hr}^{-1} \text{ ft}^{-3}\text{)}} \\ = \frac{q(t)}{5.75 \times 10^4} .$$

The required diluent volume for each batch can now be determined by solving

$$\frac{q(t)}{5.75 \times 10^4} = \sum_{n=1}^N (4 + v_{dn}) .$$

The value of $q(t)$ corresponding to the nth batch is read from Fig. E-1.

It is apparent that the largest diluent volume is required when the first batch is drained to the waste tank. Eventually the tank contains sufficient volume so that no further inert diluent is required. This behavior is shown on Fig. E-3. The total volume of diluent is 264 ft^3 , making the 30-year volume of waste plus diluent equal to 784 ft^3 . In actual practice, the total diluent volume would probably be added at the beginning of waste collection rather than in discrete steps, as shown in the figure.

Waste Tank Design

The 937 U-tubes are installed in the tank with one end open to the interior of the vault, and the other end welded into an exhaust duct leading to the stack. Air forced into the vault passes over the outside of the tank before entering the U-tubes. This design provides about 6100 ft^2 of tube cooling surface, which is about 2.5 times the calculated

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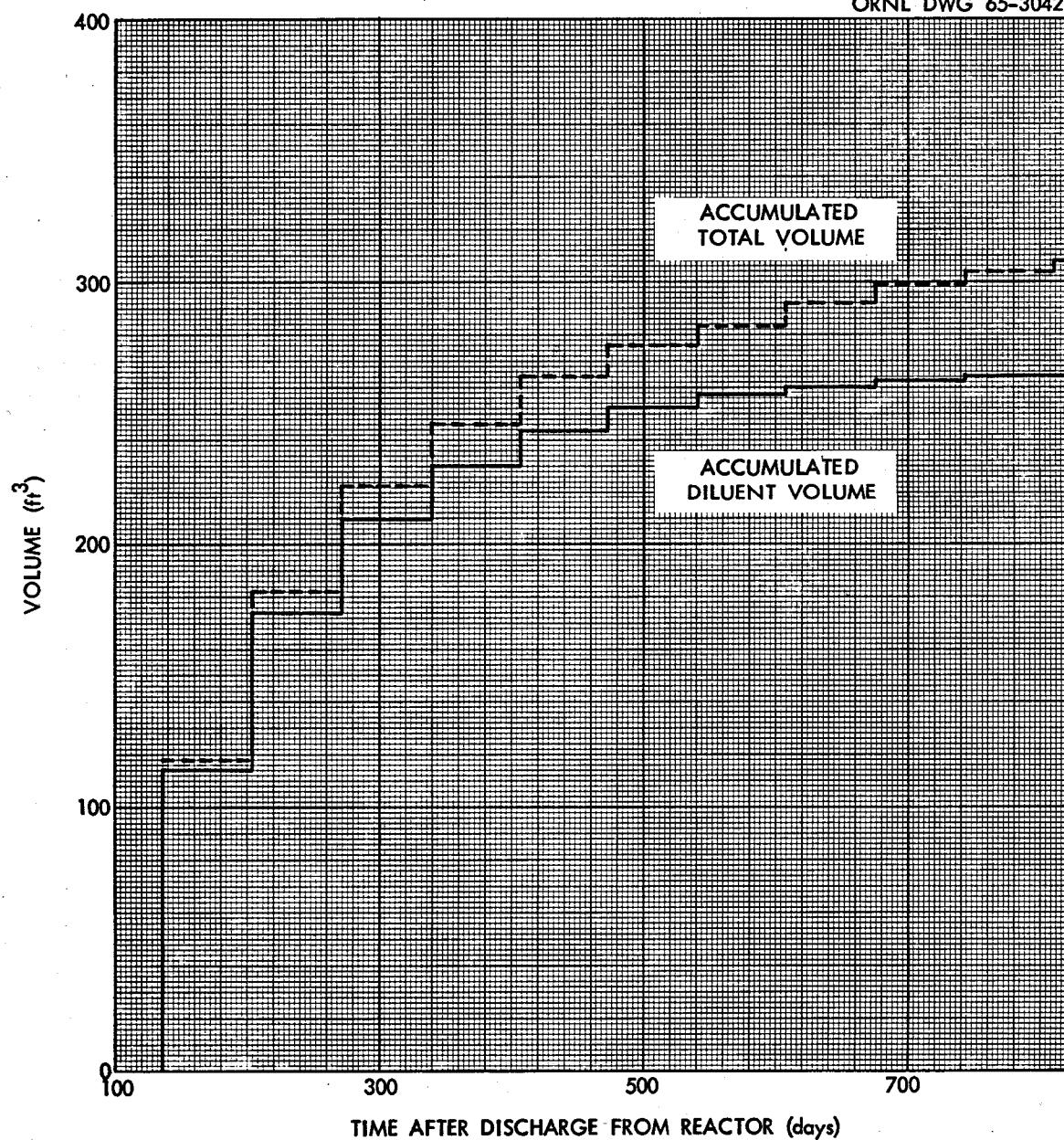


Fig. E-3. Proportions of Diluent (NaF-KF) Required to Insure Suitable Heat Transfer from Waste Tank. First batch of fuel-stream waste requires largest proportion; no dilution required after 744 days.

requirement. The relatively large number of tubes and the small waste volume lead to a tank that has a low height-to-diameter ratio. The tank is 16 ft in diameter and 6.33 ft high and has a storage volume of about 860 ft³. Monel was chosen as the structural material.

Underground Storage Facility

The underground storage area is shown on Drawing 58080-D, Appendix F. In addition to the vault for the fuel-stream waste tank, the area contains a storage vault for solid NaF and MgF₂ wastes from the UF₆ sorption step. The design of this portion of the waste system is discussed in Appendix B.

Fertile Stream Waste System

Design bases used for fertile-stream waste-storage were:

1. Thirty-year accumulation in a single waste tank.
2. Fertile stream power of 62 Mw (thermal).
3. Only one blanket volume (1783 ft³) discarded in 30 years.
4. Fission-product heat generation as shown on Figs. 3a and b.
5. Cooling by natural air convection.

It was estimated that the integrated heat generation rate at the end of the 30-year filling period would be only 5.9×10^4 Btu/hr, and, if the storage temperature is allowed to be as high as 900°F, this heat can be dissipated by about 40 ft² of cooling surface. Therefore it is only necessary to place cooling surfaces over the tank cross section in locations that shorten the path for heat conduction through the salt. Twelve 4-in.-diam pipes equally spaced over the cross section are provided to remove internal heat. Regions of salt most distant from a cooling surface might be molten during some period in tank lifetime, but this will not present a corrosion problem because cooling surfaces will always be covered with a frozen salt layer.

The storage tank is 13.5 ft in diameter and 13.5 ft high, providing about 1900 ft³ of storage volume. Stainless steel can be used in the construction because only occasionally will molten LiF-ThF₄-fission product mixture contact a metal surface. The tank is contained in an underground concrete vault as shown on Drawing 58081-C, Appendix F.

APPENDIX F. PHYSICAL-PROPERTY DATA AND DRAWINGS

This Appendix contains the following information:

- Table F-1 Thermal Data for LiF, BeF₂, Na, K, and NaK
Figure F-1 Calculated Density of MSBR Fuel Salt and LiF
Figure F-2 Calculated Density of MSBR Fertile Salt
Figure F-3a and b Vapor Pressure-Temperature Curves for Several Metal Fluorides
Figure F-4 Vapor Pressure of NaK
Figure F-5 Viscosity and Thermal Conductivity of LiF
Figure F-6 Viscosity-Temperature Curve for NaK (22.3-77.7 wt %) Alloy
Figure F-7 Properties of NaK (22.3-77.7 wt %) Alloy
Figure F-8 Process Flowsheet for Fuel and Fertile Streams
Figure F-9 Underground Storage System for Fuel-Stream Waste
Figure F-10 Underground Storage System for Fertile-Stream Waste
Figure F-11 Arrangement of Processing Equipment for Fuel and Fertile Streams

Table F-1. Thermal Data for LiF, BeF₂, Na, K, and NaK

Latent Heat of Vaporization (cal/g)

- LiF : 1690 (ref 38; calculated from vapor pressure data)
BeF₂ : 1070 (ref 11)
Na : 1038 (ref 6)
K : 496.5 (ref 6)
NaK (22.3-77.7 wt %): 617 (calculated)

Heat Capacity (cal/g °C)

-
- LiF : 0.598 (ref 37)
-

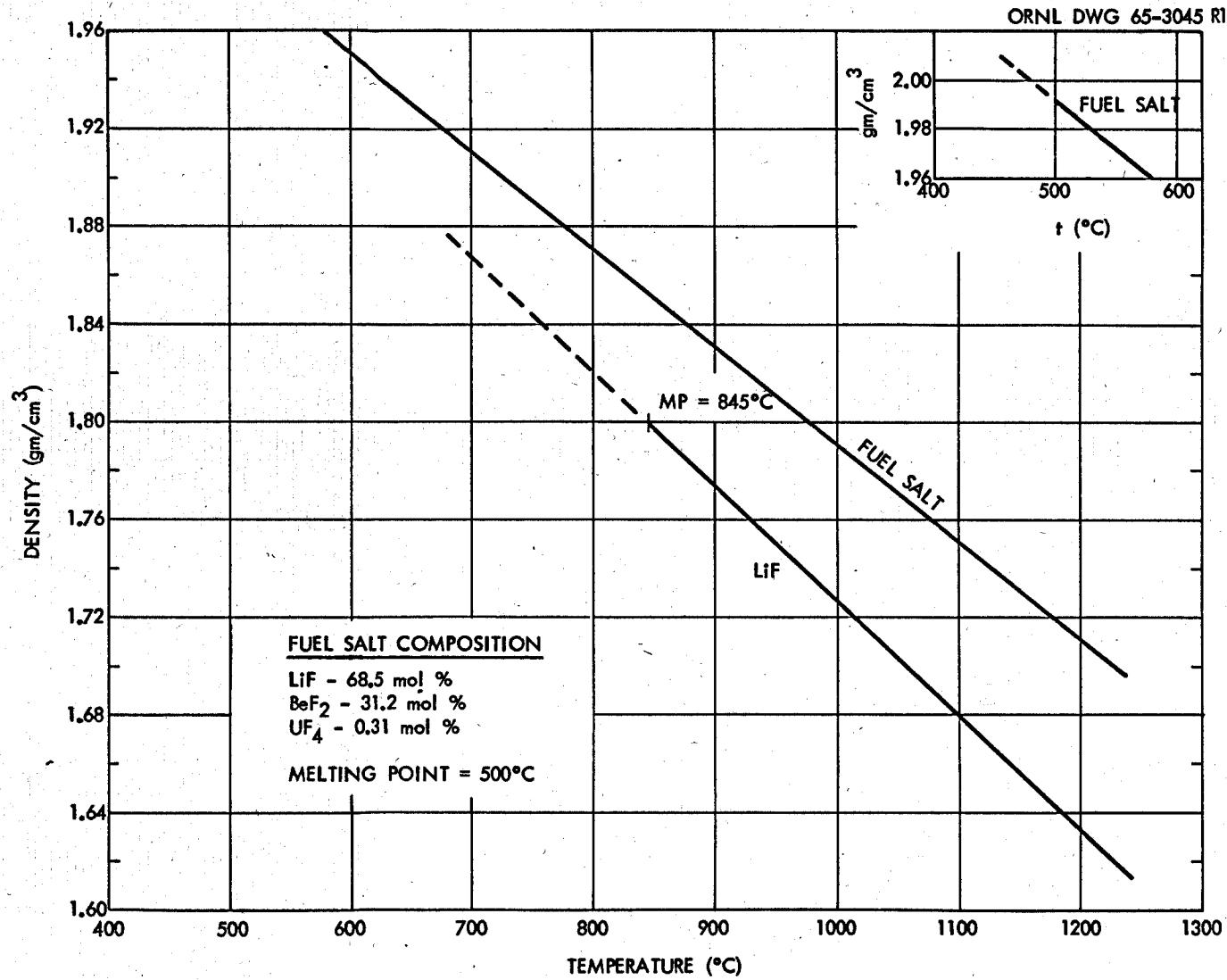


Fig. F-1. Calculated Density of MSBR Fuel Salt and LiF.

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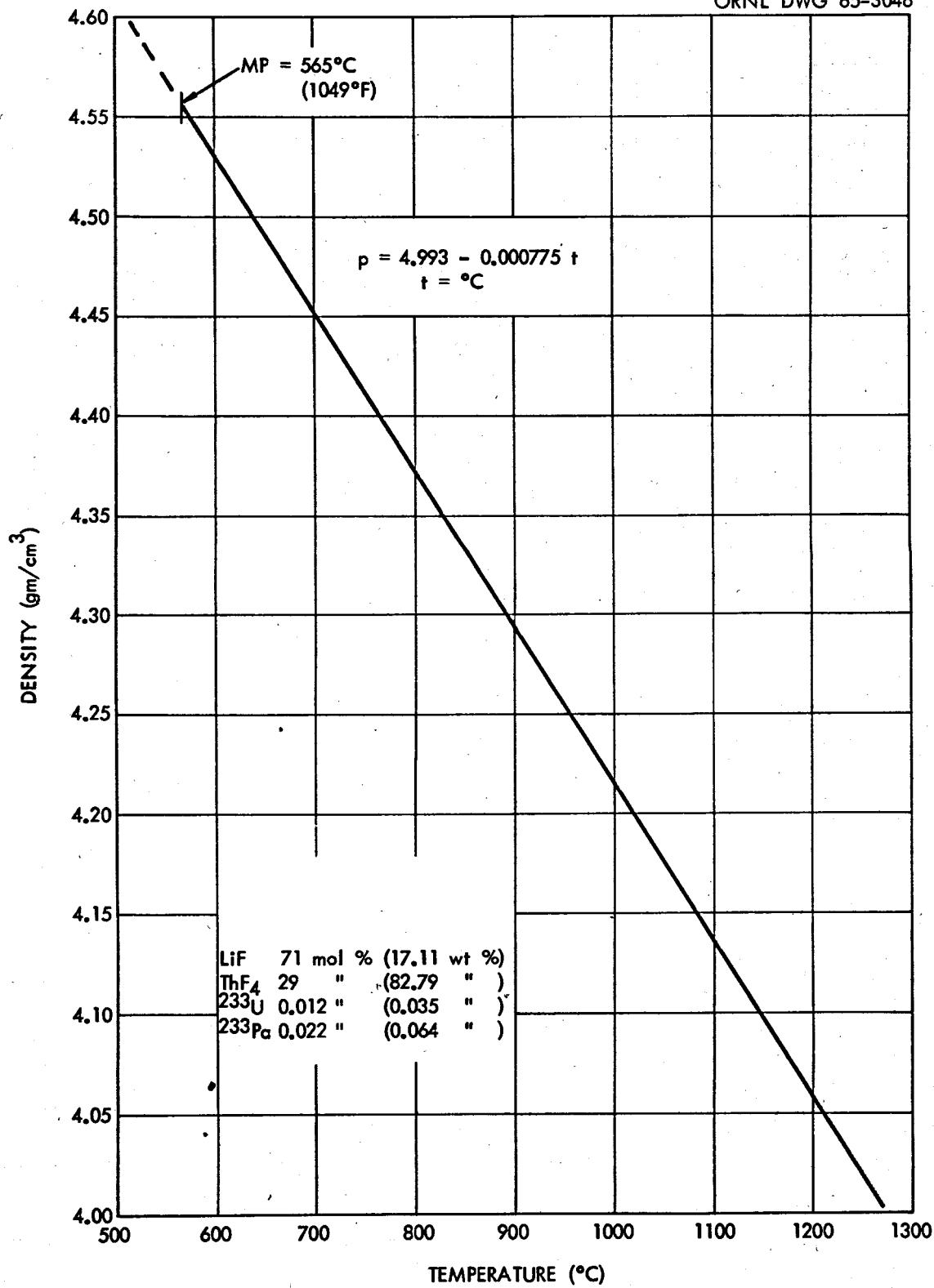


Fig. F-2. Calculated Density of MSBR Fertile Salt.

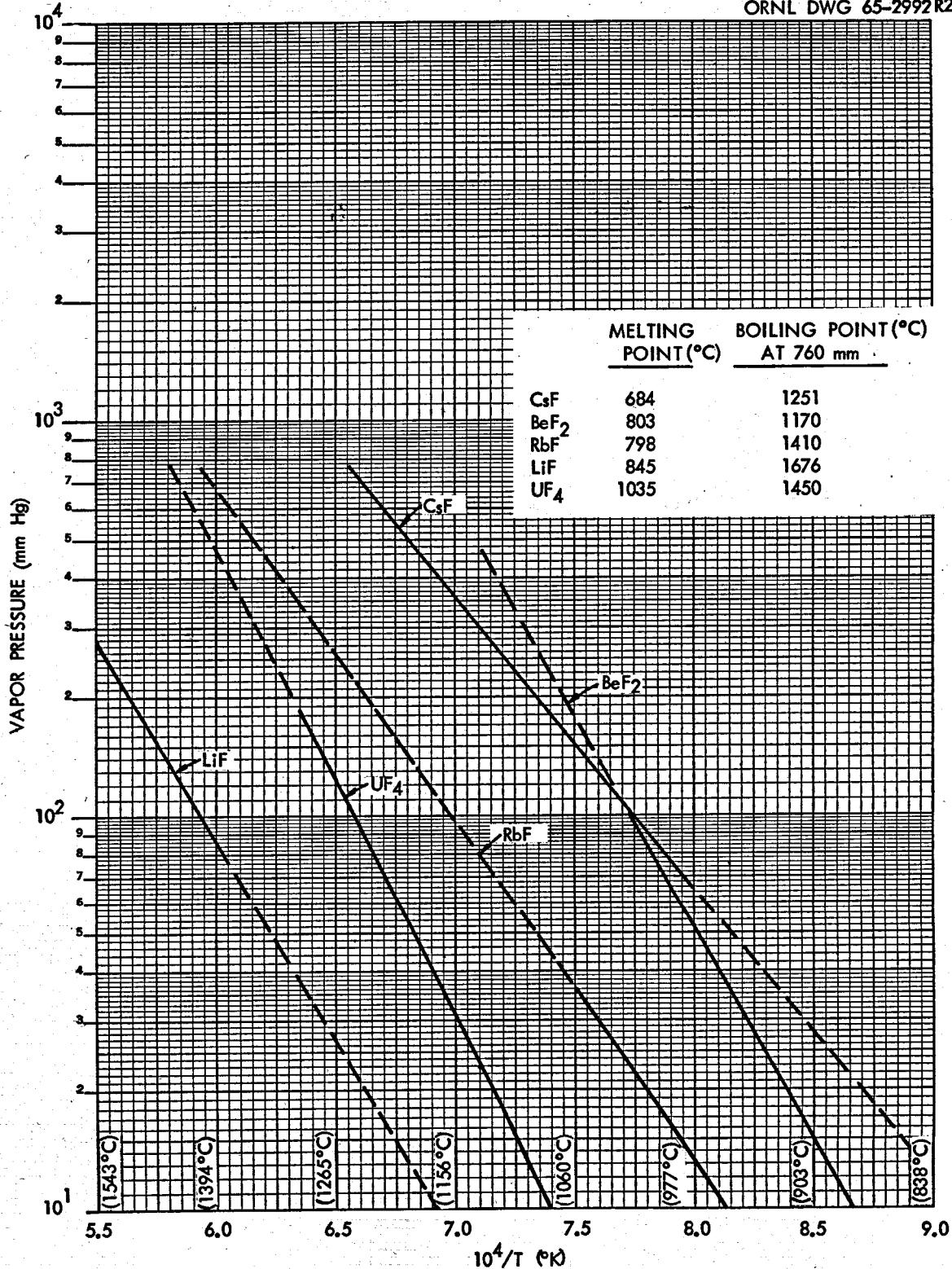


Fig. F-3a. Vapor Pressure - Temperature Curves for Several Metal Fluorides.

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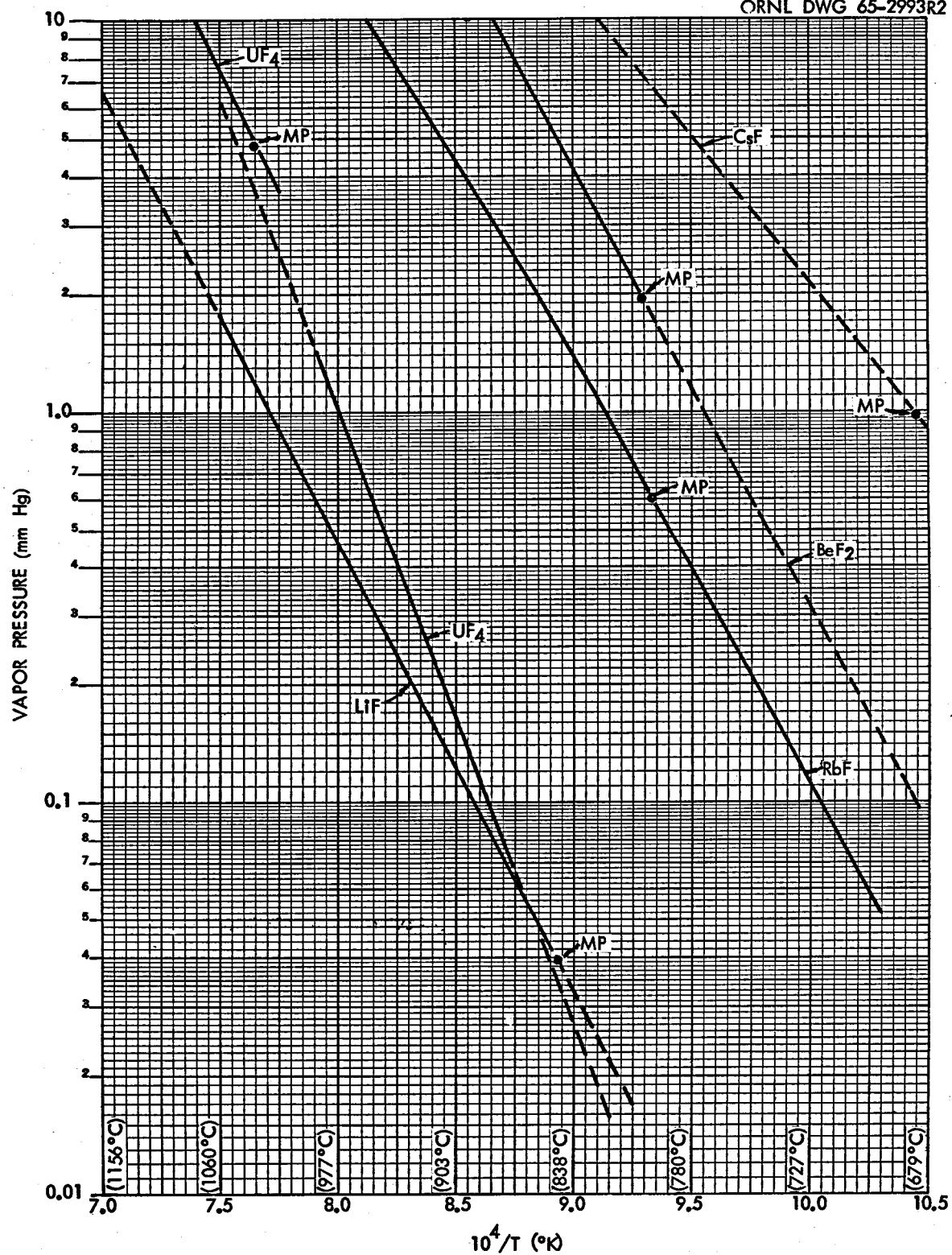


Fig. F-3b. Vapor Pressure - Temperature Curves for Several Metal Fluorides.

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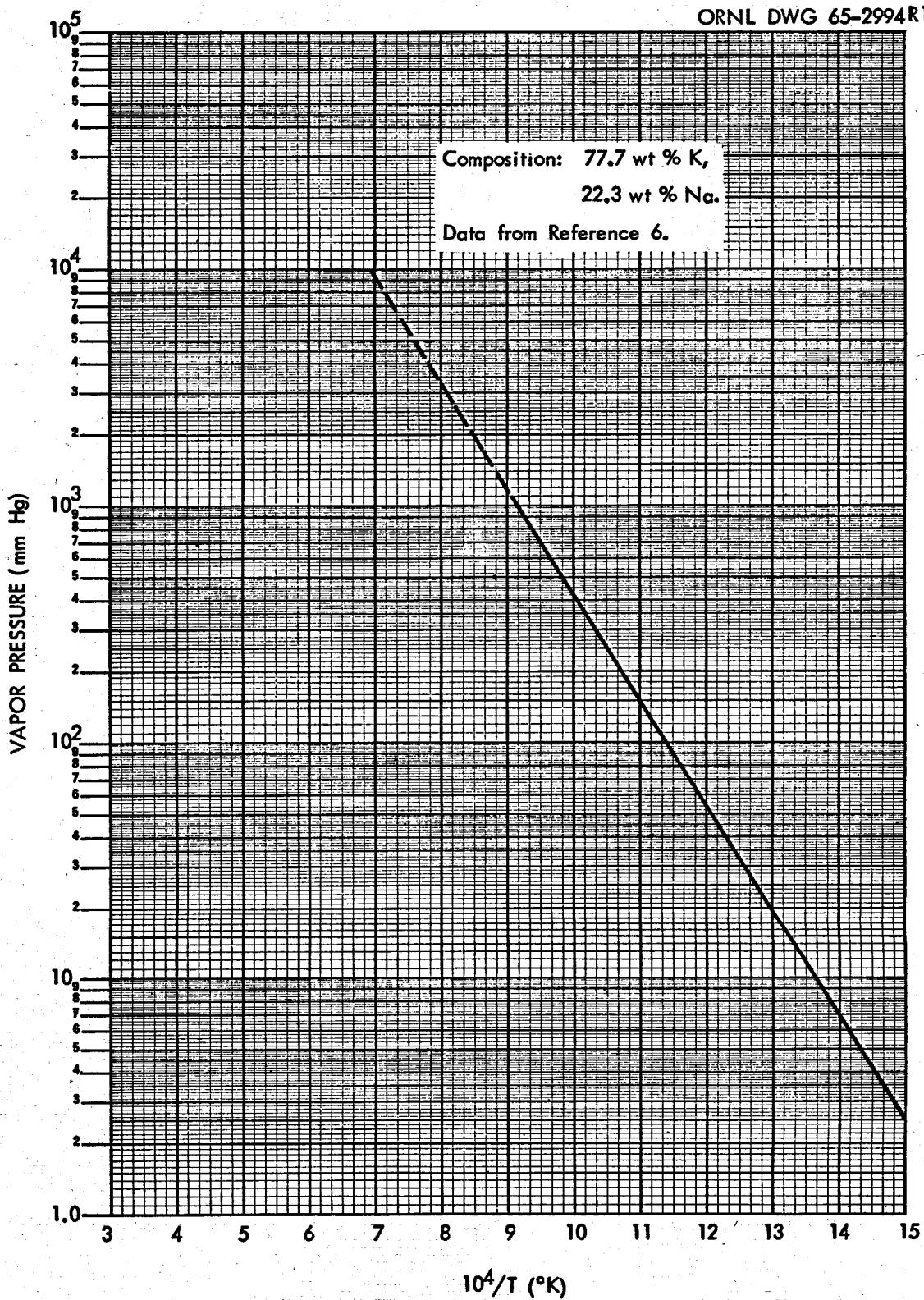


Fig. F-4. Vapor Pressure of NaK Eutectic.

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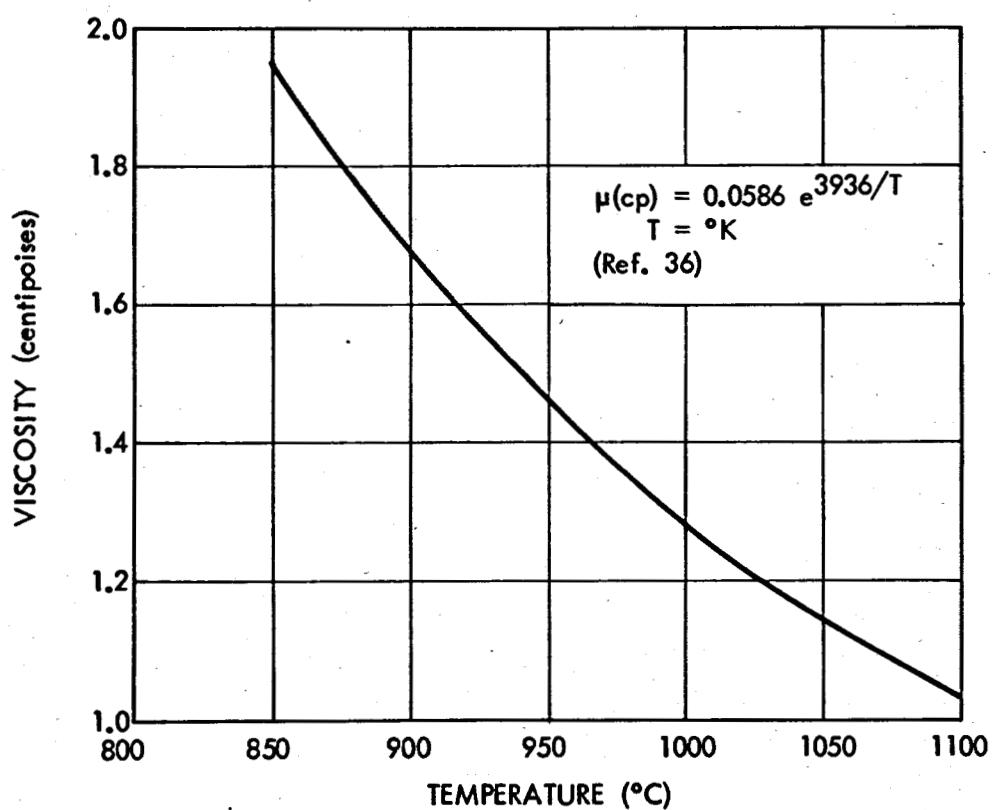
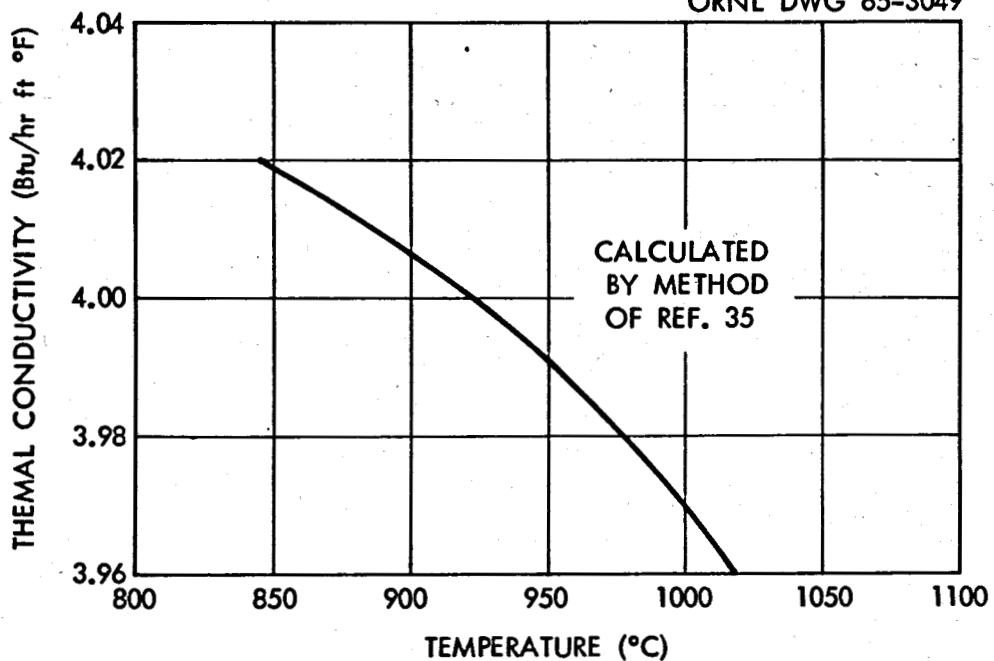


Fig. F-5. Viscosity and Thermal Conductivity of LiF.

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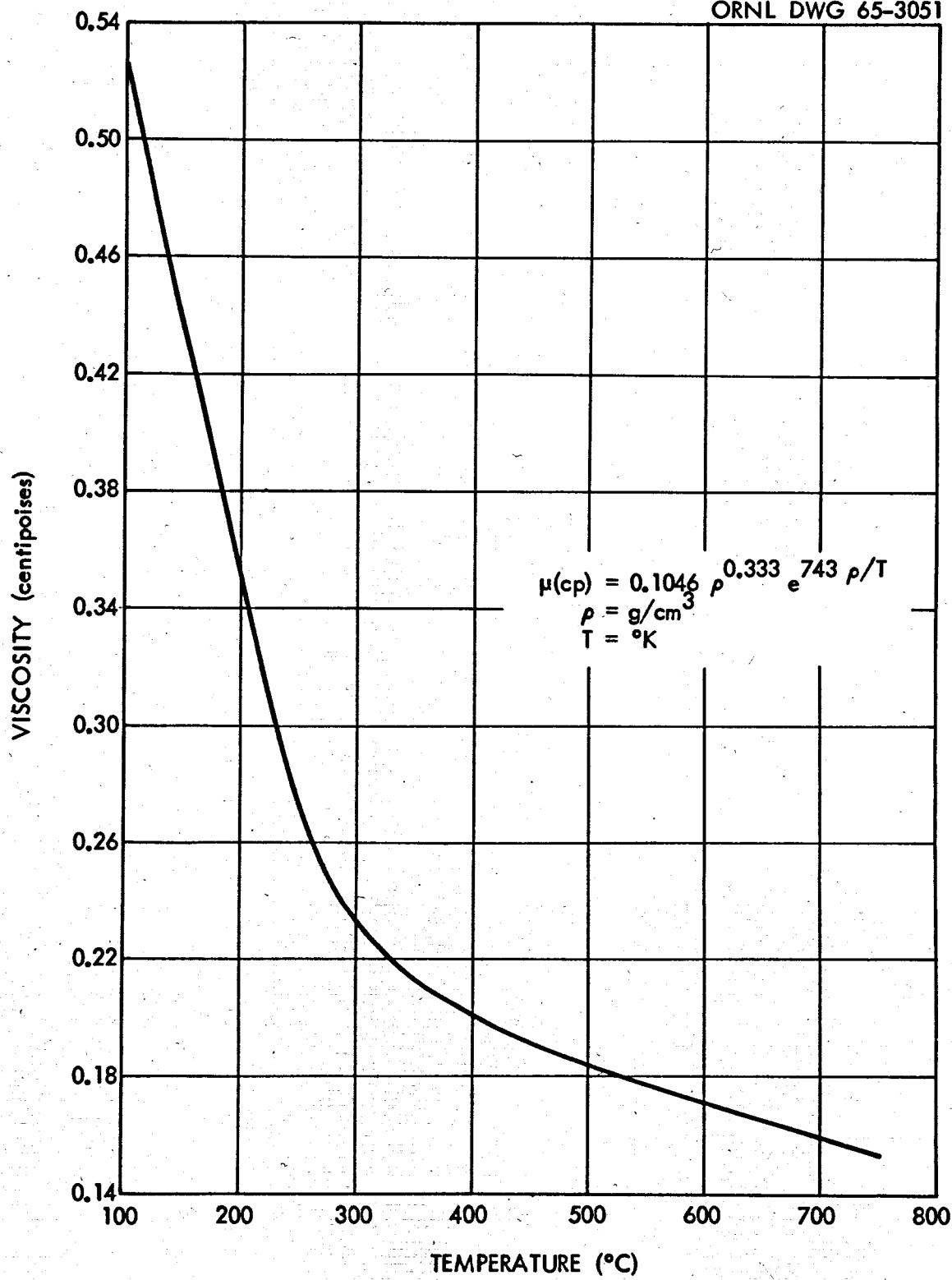


Fig. F-6. Viscosity-Temperature Curve for NaK (22.3-77.7 wt %) Alloy.
(Data from Reference 6.)

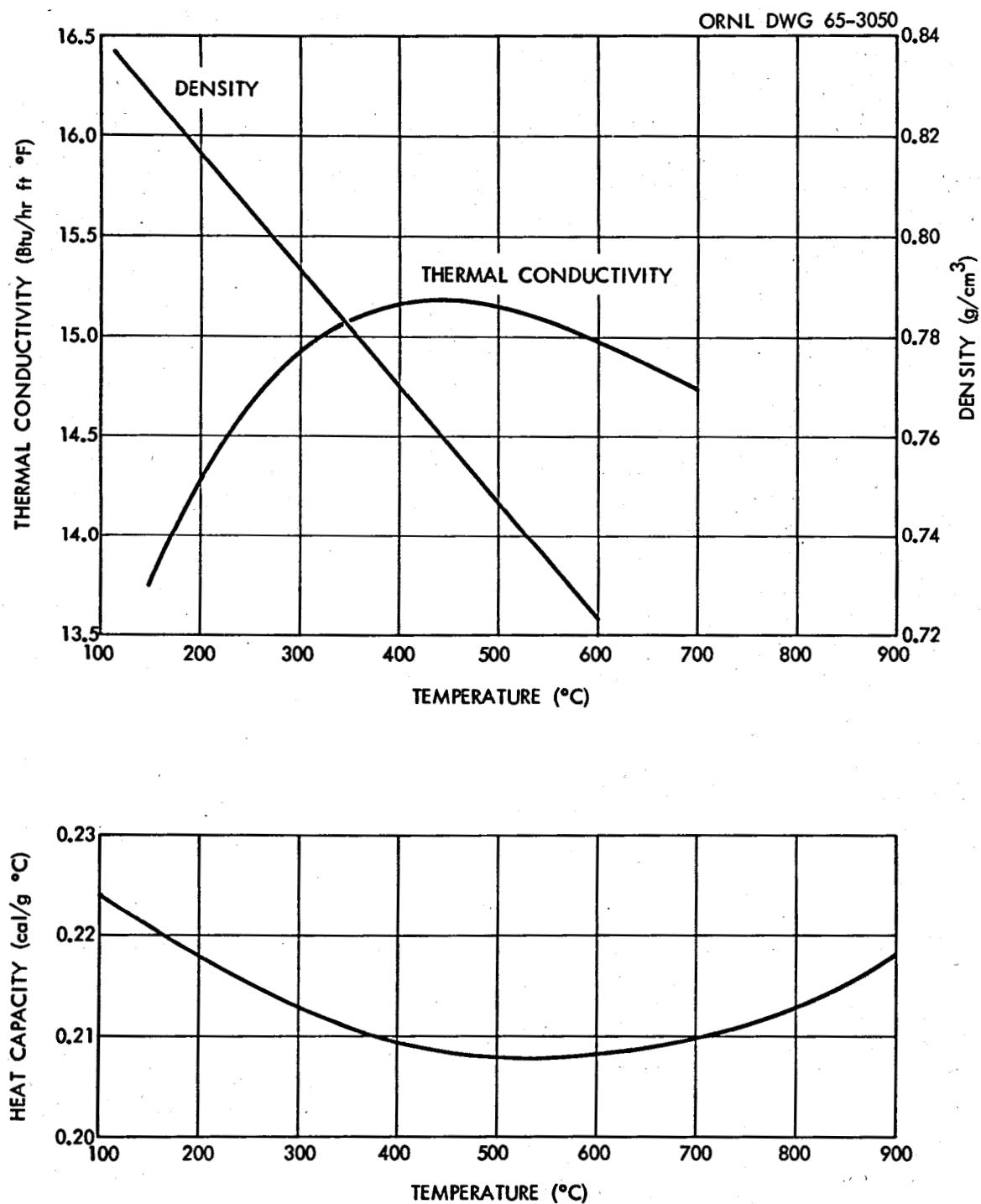


Fig. F-7. Properties of NaK (22.3-77.7 wt %) Alloy. (Data from Reference 6.)

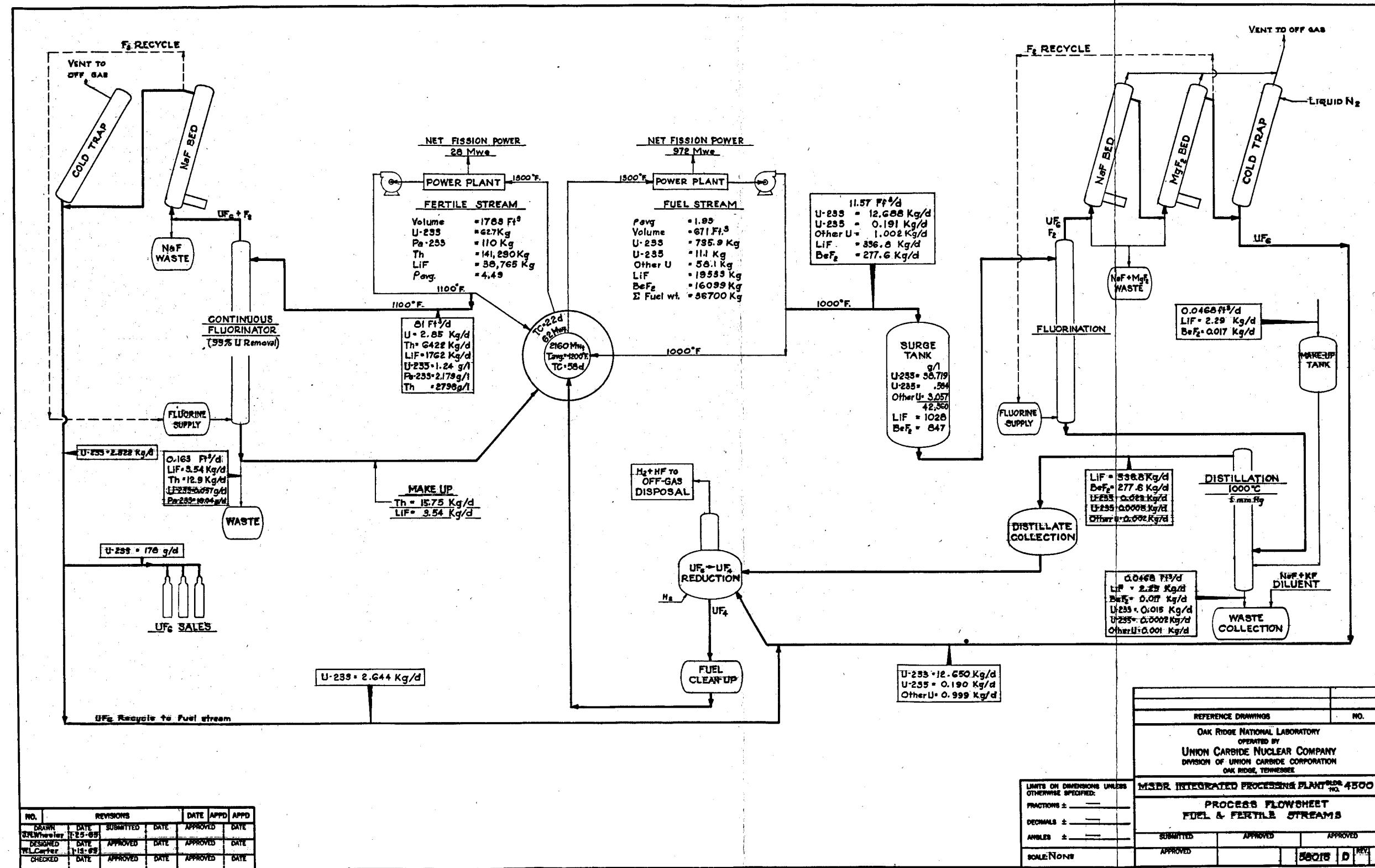


Fig. F-8. Process Flowsheet for Fuel and Fertile Streams.

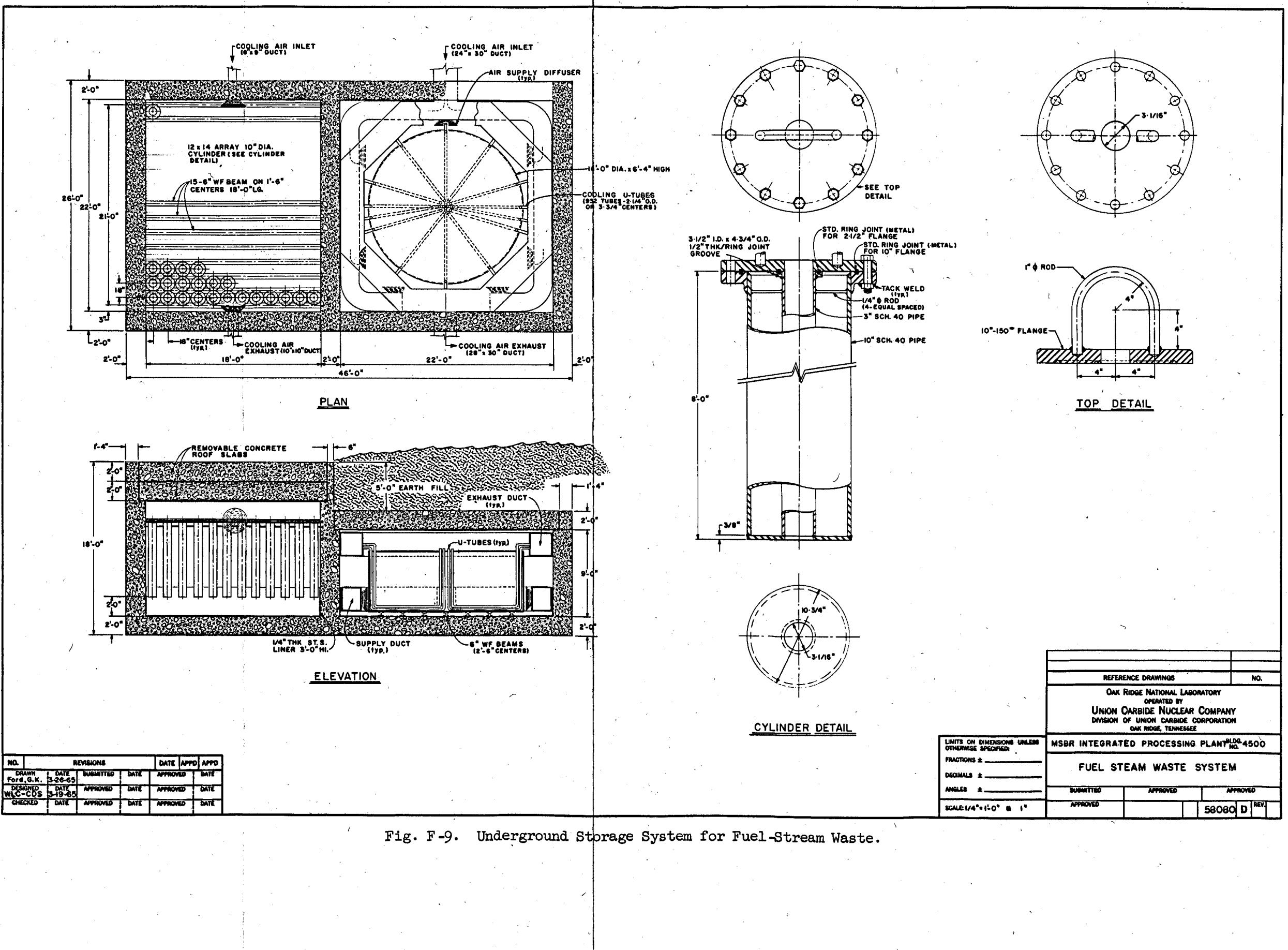


Fig. F-9. Underground Storage System for Fuel-Stream Waste.

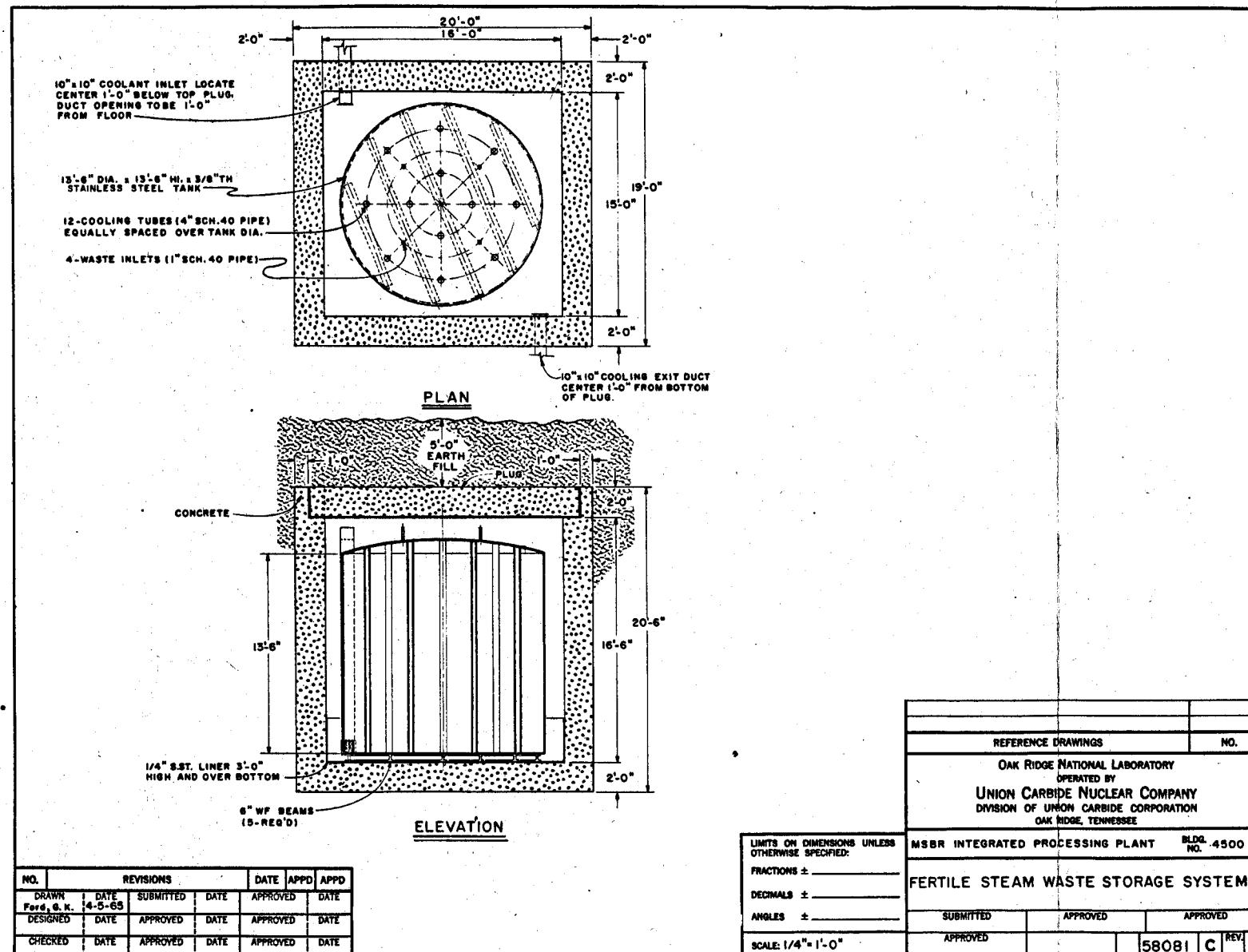


Fig. F-10. Underground Storage System for Fertile-Stream Waste.

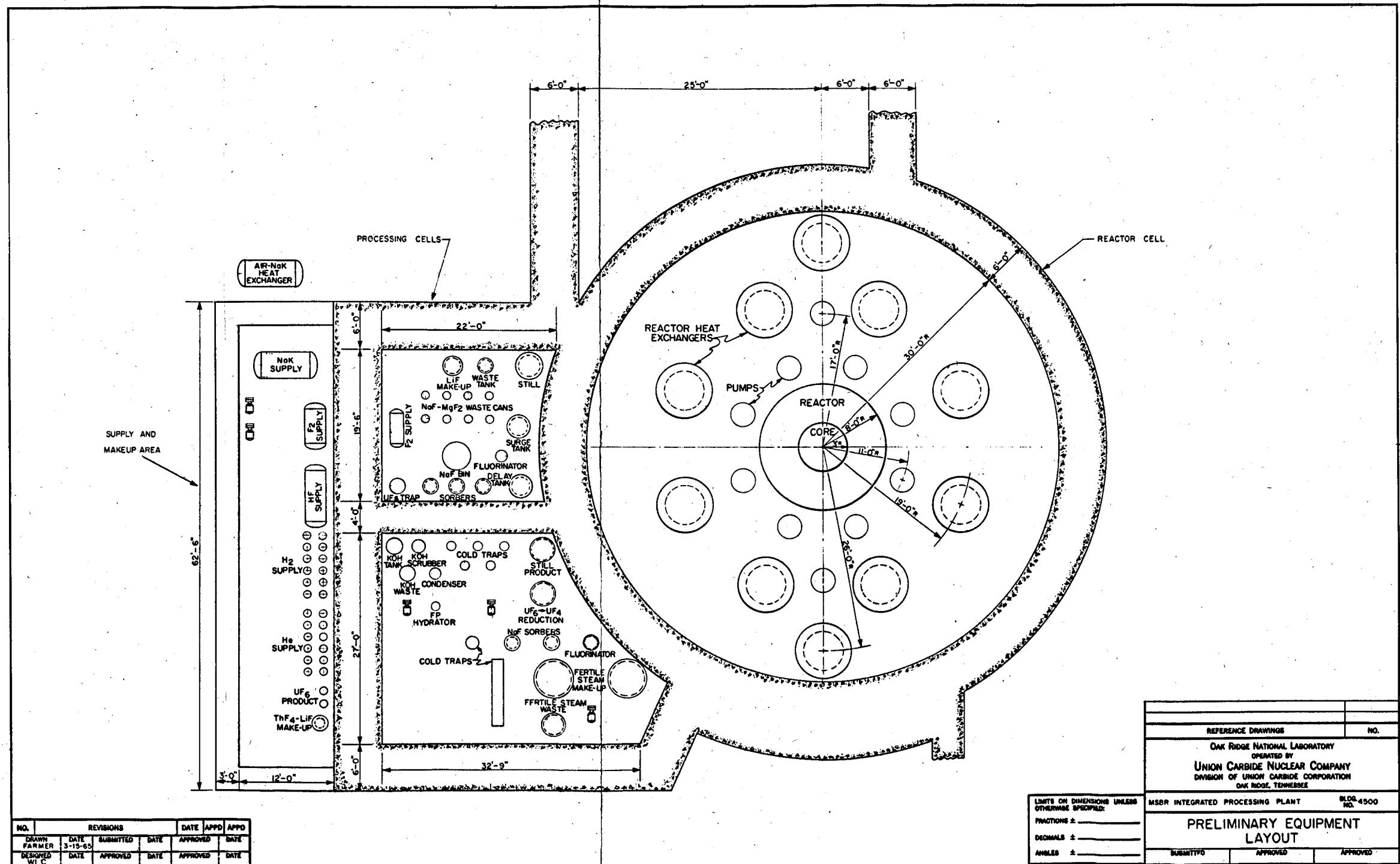


Fig. F-11. Arrangement of Processing Equipment for Fuel and Fertile Streams. The highly radioactive operations in fuel-stream processing are carried out in the smaller cell (upper left). The other cell houses equipment for the fertile stream and the cooler field stream operations.

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