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DESCRIPTION AND PROPOSED OPERATION OF THE FUEL CYCLE FACILITY FOR THE SECOND EXPERIMENTAL BREEDER REACTOR (EBR-II)

by

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Chemical Engineering Division

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FOREWORD

A large number of people* at Argonne National Laboratory have worked or are working on the EBR-II project, a project for demonstrating the engineering and economic feasibilities of a fast power breeder reactor complex. Responsibility for various major aspects of the EBR-II project has been delegated to several divisions of the Laboratory. Many reports have been issued on work performed in connection with the EBR-II project. It would not be convenient in this report to discuss the aspects of fuel recycle in the depth provided by individual reports. Rather, it is the objective of this report to integrate into a single report descriptions of various designs and operational features of the Fuel Cycle Facility. It is hoped that the References section of this report will serve the needs of those who desire more information. A number of subjects covered in this report which have not been reported elsewhere fall between major areas of investigation or represent administrative decisions, e.g., material transfer, waste handling and disposal, sampling, and criticality problems. The authors are pleased to present this report in the belief that it provides a general and useful picture of the Fuel Cycle Facility and its operations.

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*A list of the persons consulted in preparation of this report and their specialties is given in the Acknowledgment, page 185.

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ABSTRACT

The Fuel Cycle Facility for the Second Experimental Breeder Reactor (EBR-II), the process equipment, and the operations to be conducted in the facility are described. The Fuel Cycle Facility is a plant for reprocessing, by pyrometallurgical methods, the core and blanket material discharged from EBR-II. The reactor core alloy is uranium-5 percent fissium and contains about 46 w/o uranium-235. The blanket material consists of uranium-238 in which plutonium is bred. Core and blanket subassemblies contained in transfer coffins are transferred between EBR-II and the Fuel Cycle Facility, which is in an adjacent building.

The Fuel Cycle Facility consists primarily of an argon-atmosphere cell where fuel processing is done, an adjacent air-atmosphere cell where reactor subassemblies are assembled and disassembled, and an operating area (for personnel) which surrounds the two cells. Because of the high levels of activity expected, the fuel-handling-and-processing equipment is designed for remote operation.

Remote processing is accomplished with the aid of bridge cranes, electromagnetic bridge manipulators, and master-slave manipulators. Transfer ports and air locks are used in the transfer of materials and equipment into the air-atmosphere cell and between the two cells. The walls between the argon-atmosphere and air-atmosphere cells and the operating area are heavily shielded, and viewing is done through thick shielding windows.

Operations may be described in general terms as including disassembly of reactor subassemblies and their constituent fuel elements, fuel purification, refabrication of fuel elements, and reassembly of the reactor core subassemblies for reloading into the reactor. Specific operations are: fuel transfer, sodium removal from the fuel subassembly, fuel-element decanning, chopping of fuel pins, melt refining, oxidation of skull material retained in melt refining crucibles, injection casting of fuel pins, final pin fabrication, canning of fuel pins, detection of leaks in canned fuel pins, sodium bonding and bond testing, fuel-element inspection, and assembly into fuel subassemblies.

Also discussed are (1) auxiliary and alternative EBR-II fuel-cycle processes including skull reclamation, blanket, and other liquid metal processes, (2) the physical metallurgy of uranium-fissium alloys, (3) fuel movement and storage, (4) sampling and analysis of fuel and waste, (5) preparation, handling, and storage of feed materials for the processes, (6) preparation, handling, and storage of liquid, solid, and gaseous radioactive wastes. Special problems of criticality, accountability, radiation shielding, disposal of scrap fuel and unrecoverable fuel, and nitridation of fuel are discussed.

INTRODUCTION

The purpose of construction of the second Experimental Breeder Reactor (EBR-II) at the National Reactor Testing Station in Idaho is evaluation of the technical and economic feasibilities of electrical power production by fast breeder reactors.(1,2) Since EBR-II is a fast breeder reactor, its successful operation will constitute a significant forward stride in achieving a long-range atomic energy objective of complete utilization of uranium resources. Because complete analysis of reactor performance must be based on the entire fuel cycle, it was decided to build and operate a pilot-scale plant for fuel recovery and recycle adjacent to the reactor in order to demonstrate the entire fuel cycle.(3) Pyrometallurgical processes were chosen for recovery and purification of discharged fuel materials because of potential advantages of these processes,(3,4) principally simplicity and the ability of such processes to process short-cooled fuels. The latter advantage is particularly important for the high-fissionable-material-content fuels of fast reactors, because it makes possible minimum fuel inventories and, hence, minimum charges for fuel inventory. The incorporation of a pyrometallurgical processing plant in the EBR-II reactor complex also permits an evaluation of the technical and economic feasibilities of pyrometallurgical procedures.

The processing plant, which has been named the Fuel Cycle Facility,(2) provides not only for recovery and purification of fuel material, but also for the remote refabrication of new fuel elements. Thus, both the feed material and the final product of the plant is an EBR-II fuel subassembly. The EBR-II reactor system will be the first reactor system in the U.S. power demonstration program to operate on a closed fuel cycle.(3) The demonstration of a closed fuel cycle is considered very important because it is believed that the present practice of avoiding the problems of buildup of heavy isotopes (U^{236}, U^{237}, Pu^{240}, etc.) by return of fuel to an isotope-separation process cannot be continued. The problems imposed by buildup of heavy isotopes must be assessed and, if serious, must be overcome, either within the reactor cycle or by the development of reactors which can tolerate buildup of heavy isotopes.

Since pyrometallurgical processes are not processes yielding a high decontamination, fuel refabrication must be accomplished by remote operation. Consequently, the entire fuel cycle is performed behind heavy sheilding. Many new developments, both in procedures and equipment, have had to be made in order to carry out the various steps in the fuel cycle by remote operation. Only by actual plant experience can the remote operations and equipment in the fuel cycle be thoroughly and reliably tested and evaluated.

This report is concerned only with the Fuel Cycle Facility and is a two-part report. Section 1 gives a general description of the facility, including the basic auxiliary equipment (such as transfer passageways, transfer locks, and service areas and equipment). Section 2 is a description and evaluation of the overall process cycle and also includes discussions of sampling, materials handling, waste disposal, and special problems such as criticality, accountability, and radiation shielding. Special attention has been given to activities such as transfer and storage of materials which are not as exhaustively reported as the major fuel-cycle activities of fuel purification and fuel refabrication.

1. DESCRIPTION OF FUEL CYCLE FACILITY

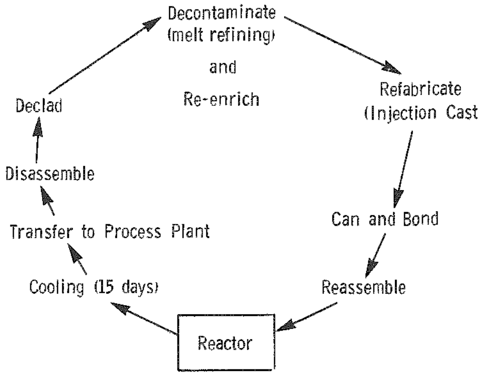
1.1 Requirements of the Fuel Cycle Facility

1.1.1 Purpose

The purpose of the Fuel Cycle Facility is to provide a plant for reprocessing, by pyrometallurgical methods, the fuel discharged from the EBR-II reactor.(3,5,6) The reprocessing (Figure 1) includes disassembly of reactor core assemblies, refining, refabrication, and reassembly of reactor core subassemblies. Although the initial effort is concerned with the fuel of the first reactor loading (partially enriched uranium-5 w/o fissium* or U-5 w/o Fs*), the long-range effort will include processing fuels of different compositions and of differing structural design. It is also a purpose of the Fuel Cycle Facility to provide a location and the services for equipment designed to demonstrate reactor-blanket processing and the recovery of fuel from slags or wastes. The locations in the Fuel Cycle Facility where these steps will be performed are shown in Figure 2.

Figure 1

SCHEMATIC DIAGRAM OF EBR-II FUEL CYCLE



An associated Laboratory and Services building provides services and equipment for examining, testing, and analysing reactor fuel which may not be essential for reprocessing purposes but which may be desirable for study of changes occurring in fuel materials under reactor irradiation.

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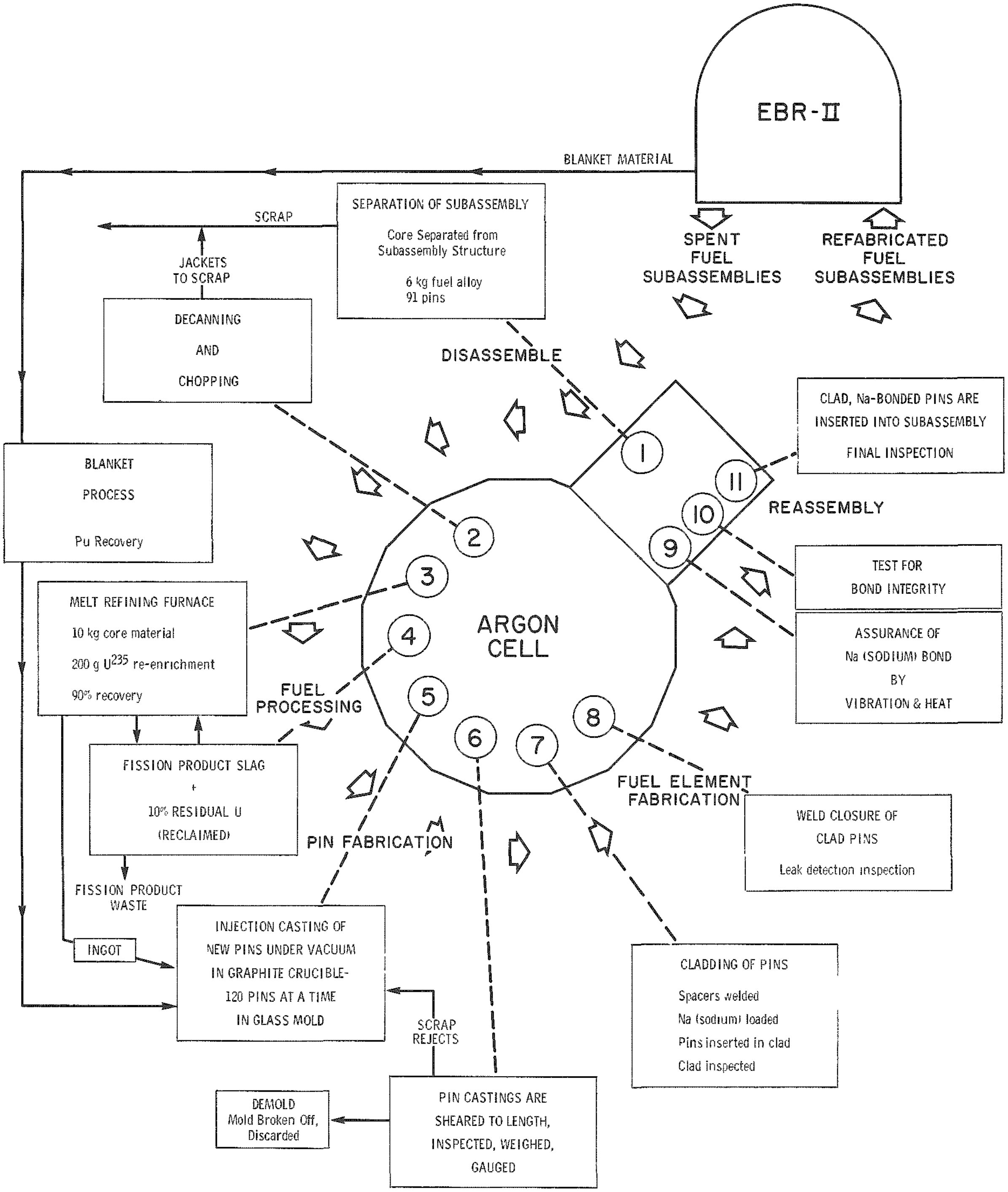
*The abbreviation for fissium is Fs, which is a mixture of relatively noble metal fission product alloying elements. Fissium in the first fuel loading consists of 2.5 w/o molybdenum, 2 w/o ruthenium, 0.26 w/o rhodium, 0.19 w/o palladium, 0.1 w/o zirconium, and 0.01 w/o niobium.

Figure 2

EBR-II FUEL CYCLE FLOW



The facility is designed to handle core subassemblies with activities up to about 500,000 c. The process cells are designed and shielded for gamma radiation levels of up to 10^6 r/hr. In general, the design is based on fuel which has undergone up to 2-a/o burnup in 135 days and which has been cooled for 15 days or longer under sodium in the subassembly-storage space within the reactor primary tank.

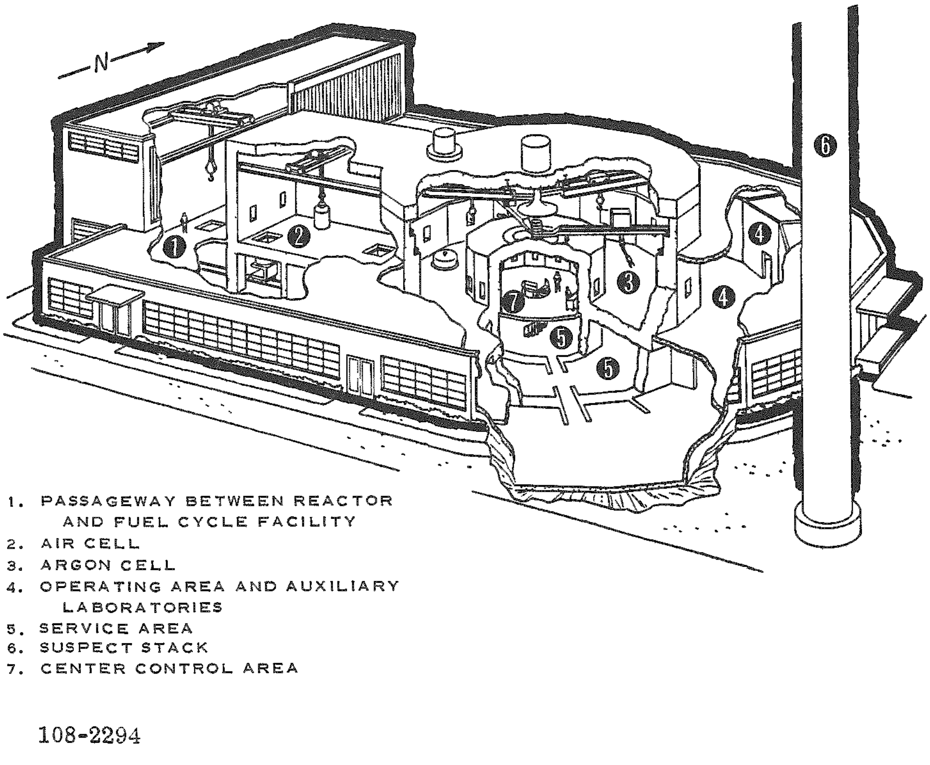
At a reactor operation of 62.5 Mwt and 2-a/o burnup, the fuel to be reprocessed would amount to about 3130 g/day of reactor operation. The first type of fuel loading will be uranium-fissium alloy consisting of about 43 a/o uranium-235, 11 a/o noble fission elements, and 46 a/o uranium-238. Other types of fuel loadings, including plutonium-containing fuels, will follow in later cycles.

1.1.2 Facilities

In order to accomplish the process functions, the following major facilities are available, as shown in Figure 3:

1) interbuilding passageway between the reactor building and the operating area at the air cell;
2) air cell;
3) argon cell (with subcell for equipment);
4) operating area surrounding the cells including auxiliary laboratories;
5) service area (service floor) beneath the operating area;
6) suspect stack, including ventilation system and fuel off-gas holdup tank;
7) center control area;
8) an analytical laboratory and service building (not shown in Figure 3).

Figure 3
CUTAWAY VIEW OF FUEL CYCLE PROCESSING FACILITY



1. PASSAGEWAY BETWEEN REACTOR AND FUEL CYCLE FACILITY
2. AIR CELL
3. ARGON CELL
4. OPERATING AREA AND AUXILIARY LABORATORIES
5. SERVICE AREA
6. SUSPECT STACK
7. CENTER CONTROL AREA

1.1.3 Required Functions

In order to reprocess the fuel from the reactor, the following steps will be performed:

1) Take fuel subassemblies from the reactor refueling machine in the reactor building, and transfer them to the air cell by means of a suitable coffin and transfer equipment. Remove adhering sodium from the subassembly.
2) Disassemble subassemblies to recover the fuel elements and blanket elements. Transfer the fuel elements to the argon cell.
3) Dismantle the fuel elements, process the fuel, reassemble the fuel elements, and transfer these to the air cell.
4) Assemble the fuel elements and necessary blanket elements into fuel subassemblies.
5) Return the fuel subassemblies to the refueling machine in the reactor building.
6) Store and handle feed materials for the processing and assembly of fuel, including new blanket elements, subassembly parts, and fuel materials.
7) Dispose of waste materials from disassembly and processing.
8) Recover fuel lost in the processing or refining operation, and return it to the process.
9) Process or dispose of blanket elements.

1.2 Fuel Cycle Loading and Operation

1.2.1 Fuel Composition and Activity

Before use, the first-cycle fuel loading will have the following approximate composition: 43 a/o U^{235}; 11 a/o noble fission elements; and 46 a/o U^{238} and other uranium isotopes. A complete fuel specification is given in Table 1.(7) The uranium-235 concentration will be maintained by adding enriched uranium. Uranium and newly prepared fuel alloy will be added to compensate for processing losses and burnup.

For 2-a/o burnup in 135 days, the fuel removed from the reactor after 15 days of cooling will have a calculated activity(8) as shown in Tables 2 through 5. The calculated fission product distribution is shown in Table 6. An estimation of fission product distribution for varying conditions of irradiation and cooling time can be obtained from a report by Burris and Dillon.(8)

Table 1

FIRST-CYCLE FUEL LOADING FOR EBR-II(7)

(5 w/o Fissium)

Element or Isotope	Atomic Weight	Atom Percent[a]	Weight Percent[a]
Uranium-235	235.12	42.88	45.64
Uranium-238[b]	238.12	46.38	49.93
Molybdenum	95.95	5.87	2.54
Ruthenium	101.10	4.33	2.00
Rhodium	102.91	0.56	0.26
Palladium	106.40	0.39	0.19
Zirconium	91.22	0.25	0.10
Niobium	92.91	0.02	0.01

[a] These do not quite add to 100.00 percent because the values are actual analytical results.

[b] Includes other uranium isotopes.

Table 2

FISSION PRODUCT RADIONUCLIDES IN DISCHARGED EBR II FUEL

Basis Burnup of 2 a/o in 135 days with 15 days of cooling of 10 kg of first-cycle fuel loading of EBR-II The fission product spectrum is based on U^{235} fission yields given in ANL-5742 (8)

Element	Mass Number	Half-life	Decay Constant (sec^{-1})	Quantity per 10 kg of Fuel[a]: g of Element	g of Radionuclides	Curies
Se	79	6.5×10^{4} y	3.7×10^{-13}	0.29	0.030	2.3×10^{-3}
Kr	85	9.4 y	2.34×10^{-9}	2.3	0.247	1.11×10^{2}
Rb	87	6.2×10^{10} y	3.55×10^{-19}	2.9	2.27	1.51×10^{-7}
Sr	89	53 d	1.51×10^{-7}	7.6	1.30	3.59×10^{4}
Sr	90	28 y	7.86×10^{-10}	7.6	3.59	5.10×10^{2}
Y	91	57 d	1.41×10^{-7}	3.8	1.79	4.51×10^{4}
Zr	93	9.5×10^{5} y	2.32×10^{-14}	24.6	4.56	1.85×10^{-2}
Zr	95	65 d	1.23×10^{-7}	24.6	2.62	5.52×10^{4}
Nb	95	35 d	2.29×10^{-7}	1.3	1.36	5.33×10^{4}
Tc	99	2.1×10^{5} y	1.05×10^{-13}	5.5	5.55	9.58×10^{-2}
Ru	103	40 d	2.01×10^{-7}	14.5	1.11	3.52×10^{4}
Ru	106	1.0 y	2.20×10^{-8}	14.5	0.388	1.31×10^{3}
Pd	107	7×10^{6} y	3.1×10^{-15}	1.5	0.271	1.28×10^{-4}
Ag	111	7.6 d	1.05×10^{-6}	0.10	0.013	2.0×10^{3}
Cd	115	43 d	1.86×10^{-7}	0.24	0.004	1.1×10^{2}
In	115	6×10^{14} y	3.7×10^{-23}	0.04	0.044	2.3×10^{-13}
Sn	117m	14 d	5.73×10^{-7}	0.36	0.004	3.2×10^{2}
Sn	119m	250 d	3.21×10^{-8}	0.36	0.045	2.0×10^{2}
Sn	125	9.4 d	8.53×10^{-7}	0.36	0.004	4.4×10^{2}
Sb	125	2.7 y	8.15×10^{-9}	0.25	0.130	1.38×10^{2}
Te	125m	58 d	1.38×10^{-7}	2.6	0.004	7.2×10^{1}
Te	127m	115 d	6.97×10^{-8}	2.6	0.029	2.58×10^{2}
Te	129m	33.5 d	2.39×10^{-7}	2.6	0.045	1.35×10^{3}
Te	132	77.7 h	2.48×10^{-6}	2.6	0.007	2.14×10^{3}
I	129	1.7×10^{7} y	1.29×10^{-15}	1.22	0.876	1.42×10^{-4}
I	131	8 d	1.00×10^{-6}	1.22	0.086	1.07×10^{4}
Xe	133	5.3 d	1.51×10^{-6}	26.0	0.077	1.42×10^{4}
Cs	135	3×10^{6} y	7.3×10^{-15}	23.2	8.01	7.05×10^{-3}
Cs	137	27 y	8.15×10^{-10}	23.2	7.66	7.41×10^{2}
Ba	140	12.8 d	6.26×10^{-7}	8.0	0.444	3.23×10^{4}
La	140	40 h	4.82×10^{-6}	7.9	0.066	3.70×10^{4}
Ce	141	33 d	2.43×10^{-7}	19.9	1.58	4.43×10^{4}
Ce	144	282 d	2.84×10^{-8}	19.9	5.00	1.60×10^{4}
Pr	143	13.7 d	5.85×10^{-7}	5.7	0.521	3.47×10^{4}
Nd	147	11.3 d	7.10×10^{-7}	22.1	0.209	1.64×10^{4}
Pm	147	2.6 y	8.46×10^{-9}	3.9	3.94	3.69×10^{3}
Sm	151	73 y	3.01×10^{-10}	2.9	0.632	2.05×10^{1}
Eu	155	1.7 y	1.29×10^{-8}	0.24	0.041	5.5×10^{1}
Eu	156	15 d	5.35×10^{-7}	0.24	0.001	5.6×10^{1}
					Total Curies	4.44×10^{5}
U	237	6.75 d	1.19×10^{-6}	negl	negl	9.0×10^{2}

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[a] The grams or curies per standard (91-fuel-element) subassembly can be obtained by multiplying the values in these columns by 0.607

Table 3

BETA AND GAMMA ENERGIES OF FISSION PRODUCT RADIONUCLIDES

Basis: Burnup of 2 a/o in 135 days and 15 days of cooling of 10 kg of the first fuel loading of EBR-II (fissionable material: U^{235})

Isotope	Total Energy per Disintegration (Mev)[a] Gamma	Beta	Gammas with Energies (Mev) over 0.75 Mev[b]	Gamma Watts	Beta Watts
Se^{79}	0	0.16		0	nil
Kr^{85}	nil	0.69		nil	0.2
Rb^{87}	0	0.273		0	nil
Sr^{89}	0	1.463		0	108
Sr^{90}, Y^{90}	nil	2.81		nil	4
Y^{91}	nil	1.55		0	144
Zr^{93}	nil	0.056		nil	nil
Zr^{95}	0.72	0.40	0.724 (49%), 0.757 (49%)	236	45
Nb^{95}	0.75	0.17	0.765 (~100%)	235	18
Tc^{99}	0	0.29		0	nil
Ru^{103}	0.57	0.20		117	15
Ru^{106}, Rh^{106}	0.32	3.25	0.88 (0.3%), 1.04 (1.7%) 1.14 (0.4%), 1.54 (0.2%)	2	9
Pd^{107}	0	0.04		0	nil
Ag^{111}	0.03	1.02		1	4
Cd^{115m}	0.02	1.61	0.95 (~2%)	nil	0.4
In^{115}	0	0.63		0	nil
Sn^{117m}	0.32	0		0.6	0
Sn^{119m}	0.89	0		0.1	0
Sn^{125}	0.09	2.27	1.9 (5%)	0.25	2
Sb^{125}	0.47	0.30		0.4	0.1
Te^{125m}	0.145	0		0.1	0
Te^{127m}, Te^{127}	0.089	0.7		0.1	0.4
Te^{129m}, Te^{129}	0.35	1.23	1.10 (~1%)	3	3
Te^{132}	0.23	0.22		3	1
I^{129}	0.039	0.15		nil	nil
I^{131}	0.40	0.57		26	13
Xe^{133}	0.081	0.345		7	10
Cs^{135}	0	0.21		0	nil
Cs^{137}	0.61	0.57		3	1
Ba^{140}	0.25	0.80		48	54
La^{140}	2.30	1.48	1.60 (~100%), 2.55 (~5%) 0.82 (~30%)	504	113
Ce^{141}	0.09	0.49		25	45
Ce^{144}, Pr^{144}	0.07	3.25	2.18 (~1%), 1.50 (~0.25%)	7	108
Pr^{143}	0	0.932		0	67
Nd^{147}	0.24	0.68		23	23
Pm^{147}	0	0.223		0	2
Sm^{151}	0.02	0.076		nil	nil
Eu^{155}	0.13	0.17		nil	nil
Eu^{156}	0.10	2.45		nil	0.4
			Total	1202	791

[a] The total gamma or beta energy per disintegration (statistical) is the weighted average energy of the various beta or gamma energies.

[b] Numbers in parentheses are percentages per decay.

Table 4

DECREASE IN FISSION PRODUCT RADIONUCLIDE HEATING WITH INCREASED COOLING TIME

Basis: 10 kg of first-cycle fuel irradiated to 2-a/o burnup in 135 days

	Watts					
	15-day Cooling		30-day Cooling		60-day Cooling	
Radionuclide	Beta	Gamma	Beta	Gamma	Beta	Gamma
Sr^{89}	108	0	89	0	59	0
Sr^{90}, Y^{90}	4	0	3.5	0	2	0
Y^{91}	144	0	120	0	82	0
Zr^{95}	45	236	39	201	28	146
Nb^{95}	18	235	17	217	14	211
Ru^{103}	15	117	12	90	7	53
Ru^{106}, Rh^{106}	2	9	2	9	2	9
Ag^{111}	4	1	1	0	0	0
Sn^{125}	2	0.25	1	0	0	0
Te^{127m}, Te^{127}	0	0.4	0	0.4	0	0.3
Te^{129m}, Te^{129}	3	3	2	2	1	1
Te^{132}	1	3	0	0	0	0
I^{131}	13	26	3	7	0	0
Xe^{133}	10	7	1	1	0	0
Cs^{137}	1	3	1	3	1	3
Ba^{140}	54	48	23	20	4	4
La^{140}	113	504	50	221	11	47
Ce^{141}	45	25	33	18	17	10
Ce^{144}, Pr^{144}	108	7	103	7	98	6
Pr^{143}	67	0	31	0	6	0
Nd^{147}	23	23	9	9	1	1
Pm^{147}	2	0	2	0	0	0

Table 5

TOTAL FISSION PRODUCT ACTIVITY AND ENERGY

Basis Burnup to 2 a/o in 135 days of first-cycle fuel loading of EBR-II

	Curies		Energy (watts)					
			Beta		Gamma		Total	
Days Cooled	6070 g fuel[a]	10 000 g fuel[b]	6070 g	10,000 g	6070 g	10,000 g	6070 g	10 000 g
15	2.7 x 10^5	4.5 x 10^5	490	800	730	1200	1220	2000
30	2.0 x 10^5	3.3 x 10^5	335	550	470	775	805	1325
60	1.3 x 10^5	2.2 x 10^5	210	340	290	480	500	820
120	0.7 x 10^5	1.2 x 10^5	135	220	145	240	280	460
240	0.3 x 10^5	0.5 x 10^5	75	125	50	80	125	205

[a] One core subassembly

[b] One melt refining charge

Table 6

FISSION PRODUCT SPECTRUM OF DISCHARGED EBR-II FUEL

Basis: Burnup to 2 a/o in 135 days with 15-day cooling of first-cycle fuel loading of EBR-II. The fission product spectra is based on U^{235} fission yields given in ANL-5742.(8)

Element	Atomic Number	Average Atomic Mass Number	Atoms Per 100 Atoms U^{235} Fissioned	Weight Percent of Fuel
Se	34	80.99	0.396	0.0029
Br	35	81	0.133	0.0013
Kr	36	85.09	2.973	0.0226
Rb	37	86.56	3.710	0.0287
Sr	38	89.60	9.525	0.0760
Y	39	89.92	4.755	0.0383
Zr	40	93.70	29.360	0.2460
Nb	41	95.00	1.580	0.0134
Mo	42	98.02	21.690	0.1900
Tc	43	99	6.200	0.0550
Ru	44	102.15	15.725	0.1440
Rh	45	103.00	2.805	0.0259
Pd	46	105.94	1.613	0.0152
Ag	47	109.26	0.100	0.0010
Cd	48	113.16	0.232	0.0023
In	49	115.00	0.042	0.0004
Sn	50	120.46	0.337	0.0036
Sb	51	123.54	0.223	0.0025
Te	52	129.23	2.209	0.0255
I	53	128.69	1.059	0.0122
Xe	54	133.76	21.775	0.2600
Cs	55	134.96	19.300	0.2320
Ba	56	138.11	6.477	0.0800
La	57	139.01	6.302	0.078
Ce	58	141.73	15.669	0.199
Pr	59	141.18	4.461	0.057
Nd	60	145.24	17.079	0.221
Pm	61	147.00	2.960	0.0386
Sm	62	149.88	2.120	0.029
Eu	63	153.34	0.174	0.0024
Gd	64	155.44	0.019	0.0002
			201.00	2.104

About 2500 g of stainless steel cladding and tubing are associated with one fuel subassembly in the core (see Section 1.2.2.1). At 2-a/o burnup of fuel, this stainless steel would have an activity of about 3 c/g or a total activity of about 7500 c. This activity is very small when compared with that of the fuel, but is of significance when consideration is given to the problems of scrap disposal (see Section 2.7.2.2).

1.2.2 Fuel and Blanket Subassemblies

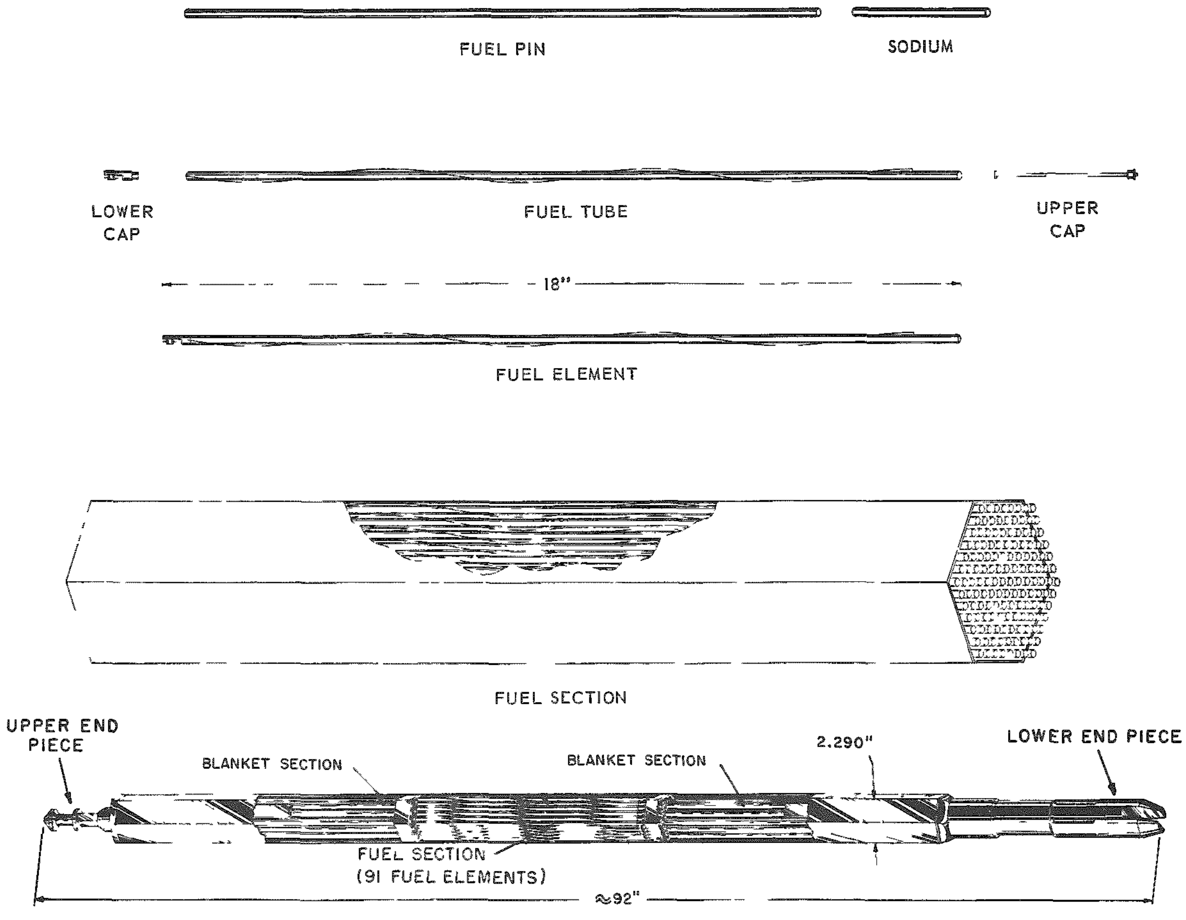
The reactor fuel and blanket material is contained in 637 hexagonal subassemblies. These include 47 core, 2 safety, 12 control, 66 inner blanket, and 510 outer blanket subassemblies.

1.2.2.1 Core Subassembly(2)

A core subassembly is 92 in. long with end pieces and consists of a hexagonal (hex) stainless steel tube, 2.29 in. across flats containing a central fuel section and two blanket sections (see Figure 4).

Figure 4

EBR-II FUEL SUBASSEMBLY AND FUEL ELEMENT



Within the central section of the hex tubing are 91 fuel elements, each of which consists of a 0.144-in.-diameter by 14.22-in.-long fuel pin in a stainless steel can.(6) The stainless steel element cans are 18 in. in overall length, 0.174 in. in diameter, and have a 0.009-in. wall. A 0.049-in.-diameter wire is spiralled around the outside of each fuel element to maintain proper spacing between adjacent elements. The assembled fuel element has a 0.006-in. annulus between the fuel and the cladding that is filled with sodium. In each end of the hex tubing are 18 blanket elements. Each blanket element consists of 0.3165-in.-diameter by 18-in.-long (two 9-in. pieces), depleted uranium blanket pins, sodium bonded to a stainless steel can. The blanket elements cans are 0.376 in. in diameter and about 21 in. long. The fuel and blanket elements are spaced on triangular lattices.

The 91 fuel pins weigh about 6070 g, and the 91 stainless steel fuel-pin cans, including lower spacer, upper spacer (or spatter cap), and external spacer wire, weigh about 1675 g. There is about one gram of bonding sodium per fuel pin. The 36 depleted uranium blanket pins weigh about 16 kg. The weights of significant components of a reactor core subassembly are tabulated in Table 7.

Table 7

WEIGHTS OF INDIVIDUAL ITEMS OF EBR-II CORE AND BLANKET FUEL SUBASSEMBLIES

ITEM		Weight (g)	
Fuel Element			
Fuel pin (14.22 x 0.144 in.)		66.7	
Sodium per fuel pin		0.8	
14½-in. stainless steel fuel element tube (cladding) around fuel pin (0.174 x 0.009 in.)		9.0	
Stainless steel spade (lower cap) and wire on fuel tube		7.1	
Top spatter cap (stainless steel)		2.4	
Total fuel element		86.0	
Core Subassembly			
Fuel - 91 fuel pins		6,070	
Inner blanket - 36 blanket pins (18 x 0.3165 in.)		16,000	
Cladding (stainless steel) per blanket pin (0.376 x 0.022 in.)		63	
Cladding - 36 blanket pins		2,268	
91 fuel elements (fuel and clad, etc.)			7,826
36 blanket elements (fuel and clad, etc.)			18,720
Hex tubing (64 in.)	stainless steel		2,570
Bottom fixture with grids and tie rod			4,060
Top fixture with grids and tie rod			1,120
Total core subassembly			34,296
Outer Blanket Subassembly			
19 blanket pins (55 in. x 0.433 in.)			48,000
Cladding (stainless steel) (190 g of stainless steel per blanket pin)			3,610
Hex tubing (64 in.)	stainless steel		2,570
Bottom fixture with grid			2,740
Top fixture			1,120
Total blanket subassembly			58,040

1.2.2.2 Control and Safety Subassemblies(2)

The control and safety subassemblies are similar to the core subassemblies except that each contains only 61 fuel elements and no end blanket elements. The hex tubing is only 1.908 in. across flats, since it is designed to operate within 2.29-in. hex tubing.

1.2.2.3 Blanket Subassemblies(2)

The blanket subassemblies are similar to the core subassemblies except that they contain nineteen 0.433-in.-diameter by 55-in.-long (five 11-in. sections), depleted uranium pins sodium bonded in stainless steel cans. The inner and outer blanket subassemblies are identical except for a difference in the channels for sodium flow in the bottom adapter.

1.3 Fuel Cycle Facility (Buildings and Equipment)

The Fuel Cycle Facility consists primarily of contiguous argon-atmosphere and air-atmosphere cells surrounded by an operating area and auxiliary laboratories. An enclosed passageway leads between this facility and the reactor. Figure 5 shows a plan view of the facility; Figure 6 is a cross section of the argon cell. The main facility is 134 ft wide by $168\frac{1}{2}$ ft long. A subcell for equipment is located below the argon cell, and a service area is located below the operating area.

Figure 5

SECTIONAL PLAN VIEW OF FUEL-PROCESSING FACILITY

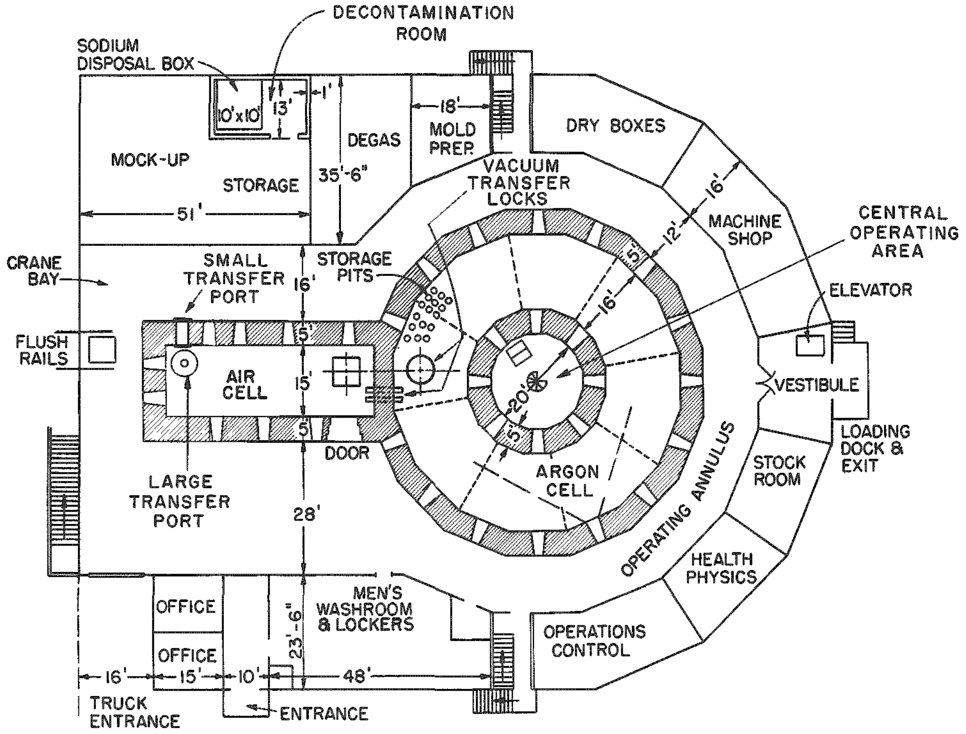
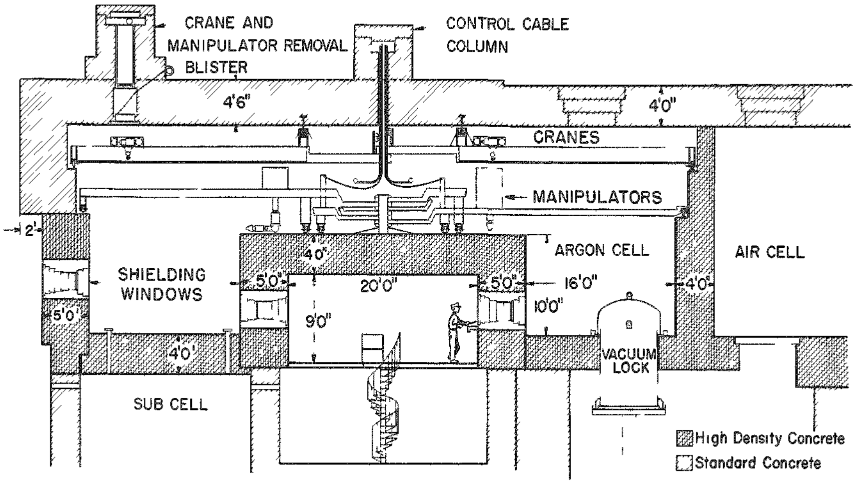


Figure 6

SECTIONAL ELEVATION VIEW OF ARGON CELL



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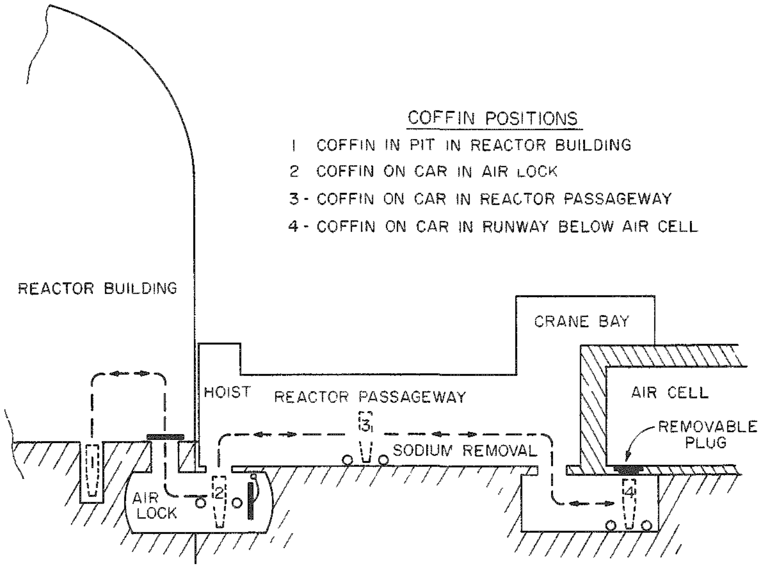
1.3.1 Reactor Passageway

The main purpose of the passageway (see Figure 7) is to provide access to the reactor building for transferring fuel subassemblies in a suitable coffin between the Fuel Cycle Facility and the reactor building. The passageway is 66 ft long and 15 ft wide. It extends from an air lock at the reactor building to the cell-operation area near the air cell. The

Figure 7

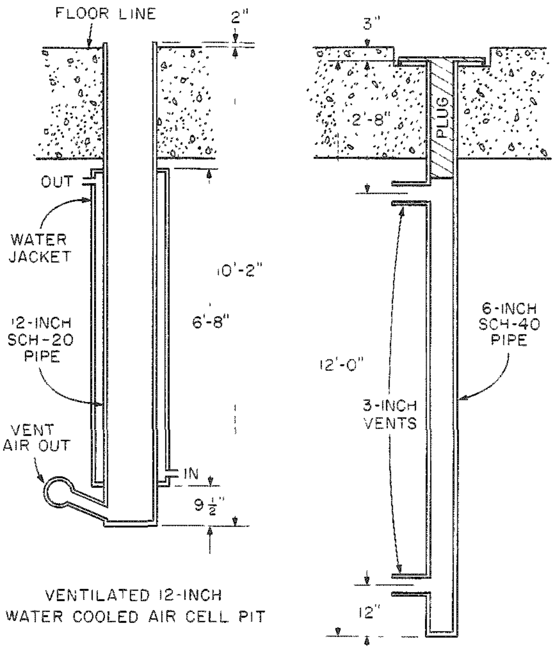
REACTOR PASSAGEWAY AND MOVEMENTS OF SUBASSEMBLY COFFIN BETWEEN REACTOR AND FUEL CYCLE FACILITY



passageway is provided with a $6\frac{1}{2}$-ft gauge railway track. An electric-powered car travels on a track in the runway which terminates below openings in the reactor building and passageway floors. The openings are closed by air-tight covers. Fuel coffins are transferred between the cars in the passageway and runway by a 20-ton hoist.

Figure 8

AIR CELL AND PASSAGEWAY STORAGE PITS



VENTILATED 12-INCH WATER COOLED AIR CELL PIT

VENTILATED 6-INCH PASSAGEWAY PIT

There are nineteen 12-ft-deep, direct-air-cooled storage pits located in the floor between the tracks in the passageway. They are provided with shielding plugs which are 2 ft long (see Figure 8).

From the passageway, a 4-in. line for liquid drain extends to a disposal system for sodium in the northwest corner of the service floor area. The drain is used in operations in which sodium coolant adhering to subassemblies is removed (see Section 2.1.2, page 60).

1.3.2 Air Cell

The main purpose of the air cell is to provide a radiation-shielded area having an air atmosphere, where subassemblies can be disassembled and reassembled and where fuel elements can be inspected and tested by remote methods. The air cell also serves as a terminal for the air locks to the argon cell and as a place for servicing argon-cell equipment and for preparing scrap for disposal.

The inside dimensions of the cell are 15 ft in width, 47 ft in length, and 21 ft in height. The air cell is attached at its east end to the argon cell (see Figure 3). The cell is constructed of high-density, reinforced concrete (220 lb/ft^3) and the walls at the operating level are 5 ft thick (see Figure 5). The cell is provided with nine shielding windows, one periscope opening, one optical peephole sleeve in the walls, two viewing plug windows in the roof, and a removable roof plug. Each window position is provided with two sleeves for master-slave manipulators and with sleeves for service leads. The cell is also provided with an entry door, which is normally sealed. For material transfer, the air cell is equipped with a large transfer port leading to the service floor below and a small transfer port with its loading side in the operating area. In addition, one large and two small transfer locks connect the air cell to the argon cell.

Ten shielded, one-foot-diameter by 10-ft-deep, direct-air-cooled or ventilated storage pits (see Figure 8) are provided in the cell floor. The pits also have water jackets for cooling. The cooling water is circulated by suction to minimize the possibility of water leakage into the pits.

One 5-ton crane and two bridge-type manipulators are provided in the cell. The manipulator bridges operate on rails along the cell walls below the crane rails. The crane can pass over the manipulators, but the manipulators cannot pass each other. The manipulator and crane rails are about 13 and 17 ft above the cell floor, respectively. The cell floor is at the same level as the operating area floor outside the cell. Processing equipment height is limited to 10 ft.

The cell is maintained at a negative pressure and is ventilated at the rate of 5000 cfm through the ventilating system and filters to the suspect vent stack.

1.3.3 Argon Cell

The primary purpose of the argon cell is to provide a radiation-shielded area where fuel can be exposed to an inert atmosphere during processing operations. Of the noble gases which are inert to the fuel, only argon and helium are of sufficient availability to use. Argon appears to be more desirable because of its relative higher density (which would minimize the leakage at pressurized seals, etc.), its availability, and lower cost.

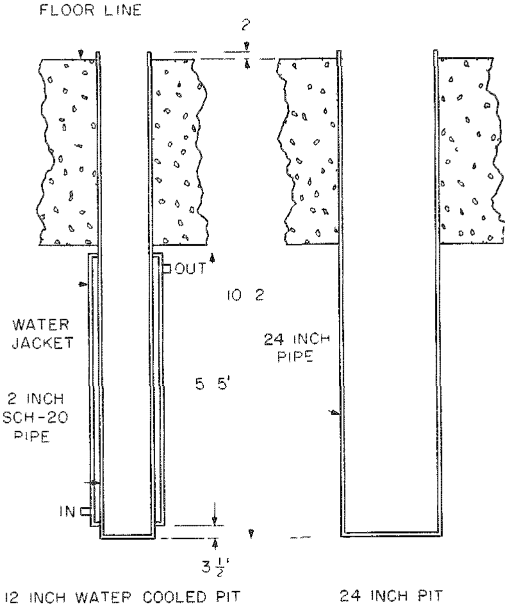
The argon cell is constructed in the shape of a 16-sided regular polygon (see Figure 5). The air cell extends out from the west side. Each of the other 15 sides is provided with a shielding window or viewing device. At the center of the argon cell is a central control area, also in the shape of a 16-sided regular polygon, with eight shielding windows (one window in alternate faces). In addition to 22 windows, the cell is provided with one periscope opening in the wall and with six viewing plugs and three periscope openings in the roof.

The distance from the center to the face of the central control area is 15 ft (30 ft in diameter). The cell floor area consists of the 16-ft-wide annular area between the two polygons and thus consists of 16 bays of trapezoidal shape which are 148 in. wide at the outer sides and which taper to 72 in. wide at the inner sides. The cell is 22 ft high inside, and has a volume of about 60,000 ft^3. The outer ceiling of the central control area extends 10 ft above the cell floor. The central control area is entered from below by a stairway from the subcell. The cell walls at the operating level are 5 ft thick and are constructed of reinforced concrete. The walls inside the cell are lined with zinc-coated sheet steel.

In the cell floor on a radial line between 14 of the bays there are 13 sets of five one-foot-diameter pipes for service feedthrough sleeves from the subcell (see Figure 11, page 48). Electrical leads for lights and window-shutter motors are of MI (mineral-insulated) cable and have been cast into the walls. There are two small transfer locks to the air cell in the wall between the cells, and one large transfer lock in the floor to a passageway which extends to a port in the air cell floor (see Figure 5, page 24).

Figure 9

ARGON CELL STORAGE PITS



The argon cell is provided with 17 one-foot-diameter, 10-ft-deep, water-cooled storage pits and two 2-ft-diameter, 10-ft-deep, noncooled storage pits (see Figure 9). The water is circulated in the pit jackets by suction to minimize the possibility of water leakage into the pits.

The argon cell is equipped with a 12-in.-diameter pipe to an emergency relief vent system. The pressure controls are designed to maintain a nominal pressure of minus 2 in. of water pressure in the cell, but a pressure range of minus 1 to minus 8 in. of water is possible. At pressures above about minus $\frac{3}{4}$ in. of water, the system will vent through the filters and ventilating system to the suspect vent stack; at pressures above about plus 1 inch of water, the system will vent directly to the atmosphere. Each vent system is equipped with a liquid seal to prevent backflow of air during normal operations. At the altitude of the facility (4900 ft above sea level), the atmospheric pressure averages 12.25 psia.

The cell is also provided with an emergency argon-supply system to maintain the cell pressure in the event that normal additions of argon are insufficient to maintain the pressure. Argon will normally be lost from the cell through the use of the transfer locks and through use of process vacuum pumps.

The cell is provided with two cranes and six manipulators. The crane and manipulator bridges pivot in the center of the cell above the central control area, and the outer ends operate on circular rails. The crane bridges operate on outer rails 17 ft above the floor and on inner rails suspended from the roof. The manipulators operate on outer rails $11\frac{1}{2}$ ft above the floor and on inner rails mounted on the central control area roof. Process equipment height is limited to 10 ft. The cranes cannot pass each other but can pass above the manipulators. The manipulators cannot pass each other.

1.3.4 Argon Cell Atmosphere

The reactive impurities in the argon atmosphere should be low enough so that excessive reaction with exposed fuel does not occur. Cooling of the cell atmosphere is necessary to remove the heat produced by the electric lights, furnaces, motors, and self-heating of the radioactive fuel.

The electrical heat load in the cell consists mainly of 72 one-kw electric lights. The heat load from furnace heaters and electric motors might average about 25 kw. The heating from 15 fuel subassemblies (at 1.2 kw each) might be as great as 18 kw. The argon atmosphere in the cell will be circulated at the rate of about 28,500 ft^3/min through two Freon-cooled heat exchangers in the subcell, which are capable of removing about 270 kw of heat from the cell. It is estimated that the average temperature of the inert atmosphere in the cell will be about 100°F.

Another relatively large heating source is that of melt refining skulls and skull oxides, both of which are highly radioactive, uranium-bearing process materials. These would normally be stored in the water-cooled storage pits. Their heat output might be as high as 50 kw, which would be largely removed by the cooling water.

The reactive impurities in the argon atmosphere will be oxygen, nitrogen, and water vapor, all of which are introduced as air leakage through seals at the windows, service feeds and transfer locks, and through miscellaneous leaks. Materials brought into the cell and impurities in the make-up argon should constitute a minor source of impurities.

Of the circulated argon about 2200 ft^3/min is passed through a filter (aluminum-encased, AEC type) and about 200 ft^3/min is passed through a purification system where oxygen is removed by combustion with hydrogen on a palladium catalyst. The water vapor is removed by adsorption in a desiccant drier which is regenerated periodically by heating. No initial provisions exist for nitrogen removal, except by dilution with make-up argon. Various possible methods of nitrogen removal other than by dilution, such as by liquefaction and rectification as well as removal by reaction with hot reactive metals (e.g., titanium at 900°C), are being considered.

The design is based on maintaining 5 ppm water vapor and 20 ppm oxygen in the cell atmosphere. Cell windows and floor service feedthrough sleeves(9a) are provided with argon-pressurized double seals, and the cell is steel lined to minimize inward leakage of air. The expected air leakage is unknown, but a rough estimate is 20 ft^3/day (this may be several orders of magnitude greater than will occur). Based on a maximum of 20-ft^3/day air leakage and 300 ft^3 of argon per day added to the cell, the calculated equilibrium nitrogen content is five percent. Studies are in progress to determine the effect of nitrogen content in fuel-processing operations.

Argon is normally added to the cell via the locks and processing equipment. When transfers are made into the argon cell, the locks are evacuated and filled with argon. The volume of each of the small locks is 20 ft^3, and the volume of the large lock is about 230 ft^3 (see Figures 5 and 6).

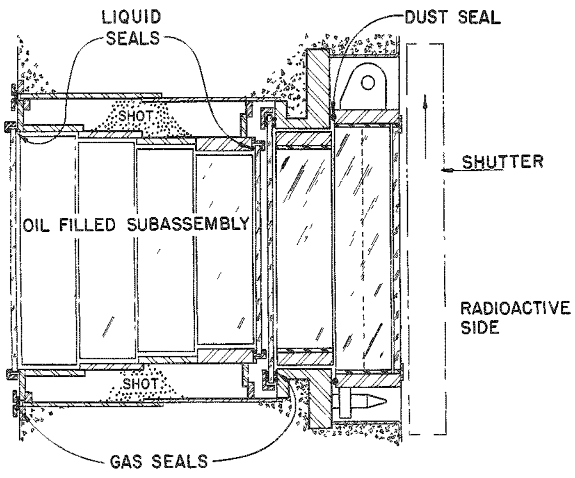
Because of the low content of water vapor in the cell atmosphere, electric-motor carbon brushes require special consideration. The heat transfer characteristics of argon are less favorable than those of air. Electric-arcing effects may be greater in argon than in air. The ionization effect of the radiation may affect the arcing potential in motors, heater coils, and other electric equipment.

1.3.5 Cell Windows and Lighting(10,11)

There are 31 windows in the argon and air cells. Each window has a total thickness of 57 in. which is made up of six 9-in. and three 1-in.-thick pieces of glass (see Figure 10). The windows are provided with argon-pressurized double seals to minimize air leakage into the cell. Two sections of glass (i.e., the sections on the radioactive side) are replaceable. The windows may be shielded with 6-in.-thick steel radiation shields (i.e., shutters) inside the cells to reduce gamma darkening when the windows are not in use. The radiation shields are suspended on tracks on which they are moved by electric motors. The total visible light-attenuation factor for the windows is about 8.4 (calculated) before exposure to gamma radiation and about 19 after exposure to 10^{10} r at the inner surface. There is a further light-attenuation factor of about 1.7 due

Figure 10

SECTIONAL PLAN VIEW OF SHIELDING WINDOW



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to eight glass-to-air surfaces. (Measurements with fabricated windows indicate the initial light transmission to be 13-14 percent.) The windows have hourglass configurations. The small inner sections of glass are about 24 in. high by 30 in. wide; the large outer sections are about 33 in. high by 36 in. wide. The angles of vision through these windows are 53° to right or left, 42° up, and 63° down with respect to the inner surfaces. The operating area floor is even with the argon- and air-cell floors, and the centers of the windows are 60 in. above the cell floor. The floor in the central control area is 3 ft below the argon-cell floor, and the centers of the windows are 30 in. above the cell floor. (Window centers are 66 in. above the floor of the control room.)

The design gamma total attenuations for the windows including buildup are approximately as follows:

Gamma Energy, Mev	Buildup Factor	Attenuation Coefficient Times Thickness	Total Attenuation
2.5	17	18.55	6.5×10^{6}
1.6	18	21.4	1.0×10^{8}
0.75	22	36.5	3.4×10^{14}

For 2-a/o burnup and 15-day cooling of the fuel, the 2.5- and 1.6-Mev gammas from 40-hr lanthanum-140 constitute about 2 and 40 percent, respectively, of the total gamma energy (see Table 3, page 20). For 10^{6} r/hr collimated gamma at the inner face, the gamma penetrating to the outer surface would be about 6 mr/hr. For 10^{6} r/hr at the inner face, due to a point source one foot away, the gamma penetrating to the outer surface would be less than one mr/hr. Because of the short half-lives of lanthanum-140 and its barium-140 precursor, these radiation levels would ordinarily be lower.

In addition to windows, the air cell is provided with one optical peephole sleeve and one periscope sleeve in the walls and two viewing plugs in the roof. The argon cell is provided with periscope sleeves in the side walls and with six viewing plugs and three periscope holes in the roof. A permanent viewing periscope is located in the operating face (replacing a window) of the argon cell.

Lighting for the air and argon cells is provided by replaceable one-kilowatt mercury vapor lamps on the inner cell walls above the windows. The argon cell has 72 lamps. The placement is as follows: three lamps on each of the 16 outer cell sides, two lamps on each inner cell side which has a window, and one on each of the eight inner cell sides without a window. The air cell has a total of 24 lamps, which are distributed as follows: nine lamps on each of the 45-ft walls and three lamps on each of the 15-ft walls.

1.3.6 Shielding(12) - Walls of Cells

At the operating level, the walls of the argon and air cells (see Figure 5) are 5 ft thick and are made of 220-lb/ft^3 concrete. The argon-cell floor and the ceiling of the central control area are 4 ft thick (see Figure 6, page 25). The walls above the operating level and the ceiling vary from 5 to 4 ft in thickness, and are made of standard concrete (about 147 lb/ft^3).

The design gamma total attenuations for the 5-ft-thick high-density concrete walls, including buildup, are approximately as follows:

Gamma Energy, Mev	Thickness for Attenuation Factor of 10, No Buildup (in.)	Buildup Factor	Total Attenuation*
2.5	6.5	23	7.7×10^7
1.7	5.4	46	2.8×10^9
0.8	3.7	300	5.4×10^{13}

*Total attenuation, I_0/I, is defined as follows:

$$I_0/I = e^{\mu x}/B_f$$

where

B_f = buildup factor
μ = attenuation factor
x = thickness of shielding.

For 10^6 r/hr collimated gamma at the inner surface, the gamma reaching the outer surface would be about 0.4 mr/hr. For 10^6 r/hr at the inner surface, due to a point source one foot away, the gamma at the outer surface would be less than 0.1 mr/hr.

1.3.7 Transfer Ports

The purpose of the transfer ports is to provide a means for transferring materials into and out of the air cell. All materials which are to be transferred into or out of the argon cell must pass through the air cell and hence through the transfer ports. The air cell is provided with one large and one small transfer port for normal communication with the operating area.

1.3.7.1 Small Transfer Port

The small transfer port (between the operating area and the air cell) consists of a 12-in.-diameter horizontal pipe through the 5-ft-thick north wall of the air cell. The port is equipped with a

20-in-long cart, extensible two-rail track, and inside and outside shielding mechanisms. The pipe center is 4 ft above the floor at the operating area side of the wall and slants downward 4° toward the cell. The extensible track also slants downward toward the cell. The extensible portion of the track moves by gravity with the cart into the cell. The cart can move into the cell 38 in. for air-cell loading or unloading and may be retracted to the operating area side by a winch-operated cable. The design load for the cart and track is 25 lb.

The cell end of the port is provided with a sliding, steel-supported lead shield, 14 in. thick, which is raised and lowered in guides over the port by cables. The outside or operating area end of the port is covered by a lead-shielded end piece which has a bottom outlet, which may be closed by a lead-shielded plug. The outer end of the port also has a straight-through outlet which may be closed by a 6-in.-thick lead door. The top of the end shield is provided with a $2\frac{1}{2}$-in.-diameter window and with a grapple. The straight-through outlet is used to insert or remove nonactive materials. The bottom outlet can be coupled with a 16-in.-OD, shielded container and can be used together with the window and grapple to move active materials, such as samples for chemical analysis.

1.3.7.2 Large Transfer Port

The large transfer port is 6 ft in diameter and is located in the floor in the northwest corner of the air cell (see Figure 5, page 24). The port cover consists of three plugs (2-in. steel laminations) nested together. The plugs are 6 in., 2 ft, and 6 ft in diameter, respectively. The outer plug (6 ft in diameter) is removable in three pieces, each 6 in. thick, to reduce the load to the crane capacity of 5 tons.

A runway (see Figure 3, page 17) at the level of the service floor extends from below the port westward under the air cell wall to below a 6-ft square opening in the operating area floor just outside the air cell. An electric-powered car operates on rails in the runway between the port and floor opening to carry coffins and containers between these points. The track gauge is 78 in., and the car is 93 in. long, 93 in. wide, and 68 in. high.

Coffins and containers are moved between the operating area and the car by means of the 20-ton crane in the high-bay section of the operating area (see Figure 3). This crane can also move the subassembly coffin to and from the car on the nearby tracks in the reactor passageway. It can also move the scrap-handling coffin to the automotive truck-loading area. Subassemblies and other materials are transferred between the coffin or containers and the air cell by a 5-ton crane located in the air cell.

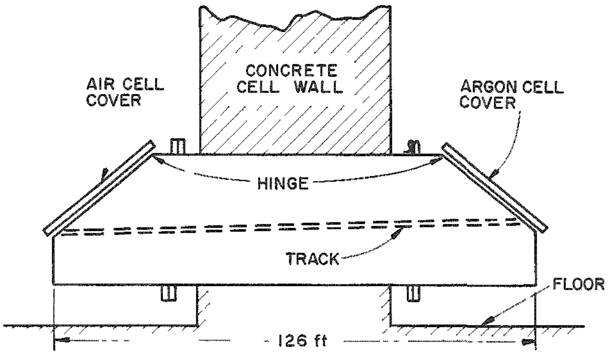
1.3.8 Transfer Locks between Air Cell and Argon Cell

The purpose of these transfer locks is to provide means for transferring materials and equipment into and out of the argon cell. All locks into the argon cell communicate with the air cell to provide the necessary shielding and to prevent spread of contamination (particularly atmospheric).

One large and two small transfer locks are provided. The small locks will be used for the frequent transfer of small items, such as fuel elements, fuel element cans, crucibles, fume traps, and samples. These may be operated relatively rapidly and have a smaller volume; thus, their use will result in smaller argon losses. The large lock will be used for less frequent transfer of large equipment and those items too large for the smaller locks.

1.3.8.1 Small Locks

The two small locks extend through the wall between the argon and air cells as shown below. Each lock is 13 in. wide by 23 in. high by $10\frac{1}{2}$ ft long and has a volume of about 20 ft^3.



The sealing covers at each end are slanted at an angle of about 35° with the horizontal and are hinged at their upper ends. They are raised and lowered by cables operated by electric-motor drives mounted on the cell walls. Each cover is provided with a rectangular rubber gasket which is stiffened by a steel inset. The openings are 13 in. wide by about 16 in. long. The covers are constructed of steel, several inches thick, and their weight provides the gasket sealing force.

A two-rail, 124-in.-long track extends through each lock at about the level of the lower edge of the lid openings. The track assembly is supported at each end by means of brackets on which the cross members rest. The cross members can be positioned in the brackets, by the manipulators, to slant the track in either direction. (The slope is $1\frac{1}{4}$ in. in 124 in.) Thus, either lock can be adjusted for cart travel in either direction.

The track is removable by raising and pulling at one end. This causes the cross member at the opposite end to drop off the highest step and fall to the bottom of the lock. The track may then be pulled out of the lock.

The standard carts which travel on the rails consist of aluminum alloy frames with a wheel at each corner. Theyare $12\frac{1}{4}$ in. in overall width and $11\frac{3}{8}$ in. in overall length. The wheels on one side are provided with 90° V-grooves for following the rail; the wheels on the other side are flat.

The containers can be up to about $9\frac{3}{8}$ in. in diameter, and can extend above and below the carts about 10 in. (20 in. overall). The basic cart frame can be altered to accommodate other types of containers. The cart frames are provided with ears on their sides for lifting in and out of the lock openings. The standard carts are carried in horizontal position to and from the locks by the manipulator, and may be set down directly on the tracks. Special carts are used for fuel-element magazines which are too long (24 in.) to be set in the locks in a horizontal position. They are carried by one end in a vertical position, and, as they are lowered into the lock, the lower wheels contact the track first and the cart rolls into the lock and assumes a horizontal position when the other wheels touch the track. Maximum weights of 70 lb, including cart, container, and load, can be accommodated. The motors which open the lock covers have interlock controls so that the covers on opposite ends of the locks are not inadvertently both opened at the same time.

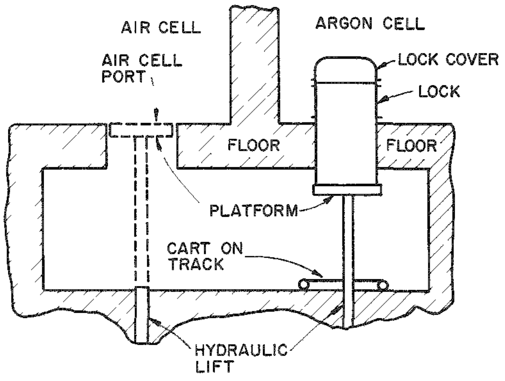
Fuel elements in magazines, pin cans, melt refining and injection casting crucibles, pin molds, pallets, samples, fuel charge materials, and crucible holders can be transferred through the small locks. Melt refining fume traps may require special carriage arrangements for small lock transfer. Scrap in cans may require large lock transfer.

1.3.8.2 Large Lock

A large lock (see sketch on next page) extends through the floor of the argon cell into a transfer cell or runway at the level of the service floor. The runway extends from below the argon cell to below a port in the air-cell floor (which is usually open). The runway is provided with removable shielding for personnel entry. The port in the air-cell floor is also the vent through which the air cell is vented to the stack through the runway. A cart is operated on tracks in the runway.

One part of the lock is a hollow, steel spool piece, 6 ft in inside diameter by 51 in. long, extending through the 42-in-thick floor of the argon cell. The top of the spool piece extends 7 in. above the floor, and its face has a groove 75 in. in diameter and $\frac{3}{8}$ in. deep for a metallic sealing gasket. A second spool piece, 33 in. long with gasket grooves, is semipermanently attached to the first one. It extends about 40 in. above the cell floor. A 6-ft-diameter by about 15-in.-high dished cover is used to close the upper end of the lock within the argon cell. The cover is hinged to the spool piece and is raised by an electric motor.

The bottom of the first spool piece extends 2 in. below the lower side of the floor and is faced for sealing with a rubber gasket on a hydraulically lifted platform. The platform is carried between the lock and the air-cell port by a car on the tracks in the runway. One hydraulic lift is located beneath the air lock in the argon-cell floor, and another hydraulic lift is located beneath the port in the air-cell floor (see sketch).



The maximum load weight is 10,000 lb; the maximum load height is 8 ft. The horizontal dimensions are limited by the 6-ft inside diameter of the spool piece and the allowance for clearance. The volume of the space between the lock cover and the platform is about 230 ft^3.

1.3.9 Suspect (Vent) Stack and Holdup Tank

The purpose of the suspect stack (see Figure 3, page 17) is to provide a high point of discharge to the atmosphere of all ventilation atmosphere suspected of containing radioactive contamination. The purpose of the holdup tank located near the northeast corner of the Fuel Cycle Facility is to provide a place of storage of active gases and vapors from the operations of melt refining and possibly of skull oxidation to await favorable meteorological conditions for release to the stack. As a secondary effect, some decay of activity can take place in the holdup tank. Xenon-133 and possibly iodine-131 are the chief nuclides to undergo decay.

The suspect stack is 5 ft in diameter and 200 ft high. It is glass-lined and is provided with two 71,000-cfm fans to draw ventilation air from the Fuel Cycle Facility, the Reactor building, and the Laboratory and Services building. The quantities of air from these locations are as follows:

Location	Pipe Size (in.)	Quantity of Air (cfm)
Fuel Cycle Facility	42	32,800
Laboratory and Services Bldg.	36	18,200
Reactor Building	30	6,900

The ventilation air is filtered before being discharged to the stack. Gases from process operations are filtered through high-efficiency filters and, in most cases, are further processed (e.g., through a tower for iodine removal) before being released to the stack. Each of the locations has booster fans and filters. The Fuel Cycle Facility has two 38,000-cfm main booster fans. The air flow in the ductwork, filters, and piping is controlled at a negative pressure.

The holdup tank, one of a series of items used to handle off-gases (see pages 164 and 165), is about 6 ft in diameter and 19 ft long, and has a volume of about 500 ft^3. Located outside the building, where additional shielding can be provided, the tank is used to receive vent gases from the melt refining (see Section 2.1.6) and possibly other operations. Normally, the holdup tank is maintained at between about 4 to 10 psia (local atmospheric pressure averages about 12.25 psia). When meteorological conditions are favorable for gas release from the stack, the tank is pumped down to about 4 or 5 psia.

The tank, after pumpdown, can receive vent gases from the melt refining operation and possibly the skull oxidation operation (see Section 2.1.7) until the pressure has increased to about 11 psia. The tank can thus receive about 250 ft^3 of gas at local atmospheric pressure in operating between the low and high pressures. This is equal to about fifteen times the volume of the melt refining bell jar.

It is anticipated that the bell jar will be evacuated three times per melt refining operation. Thus, favorable conditions for gas release would need occur only once per five melt refining operations.

1.3.10 Manipulators and Cranes

The air and argon cells are provided with three types of material-handling equipment in addition to the port and air-lock equipment. They are bridge cranes, electromechanical bridge manipulators, and master-slave manipulators.

1.3.10.1 Air Cell Crane

The air cell is provided with one 5-ton crane with a linear bridge which operates on parallel rails along the sides of the cell. The rails are 17 ft above the floor, and the crane moves above the manipulators. The bridge has four wheels (two on each side) of which one on each side is a drive wheel. The trolley which operates on the bridge has four wheels and carries a cable-operated block hoist. The trolley can be removed from the bridge through the top hatch of the cell for repair or replacement.

The crane bridge, trolley, and hoist motors are 220-v, 60-cycle AC, one-speed reversing motors. The motors are shielded by approximately 4 in. of lead at the bottoms and 2 in. of lead at the sides, and are removable. Electric power is fed to the bridge by 11 busbars on the ceiling and to the trolley by seven busbars on the bridge. The busbars used are as follows:

1 - common
2 - bridge motor
2 - lights on bridge
2 - trolley motor
2 - hoist motor
2 - three limit switches

Where no shielding is provided, radiation-resistant lubricants are used. There are no brakes on the carriage or bridge drives, but there is a mechanical brake on the hoist. The trolley and bridge drives are provided with automatic disconnect clutches so that in case of failure of the drives, the bridges and carriages can be moved by other equipment.

1.3.10.2 Argon Cell Cranes

The argon cell is provided with two 5-ton cranes (see Figures 3 and 6, pp. 17 and 25) with rotary bridges which pivot at the center of the cell. The outer end of the bridges ride on rails which are 17 ft above the cell floor; the inner end of the bridges are suspended from rails which are secured to the cell ceiling. As in the air cell, the cranes move above the manipulators. The cranes cannot pass each other. Each bridge has four wheels on the inner track and two on the outer track. Each outer wheel has a drive motor. The trolleys which operate on the bridges have four wheels and carry cable-operated block hoists. The trolleys can be removed from the bridges and lowered to the cell floor by a special motor-powered chain hoist, which is housed in a sealed can on the cell roof, which is called a blister.

The crane bridge, trolley, and hoist motors are 220-v, 60-cycle AC, one-speed reversing motors. The motors are shielded by approximately 4 in. of lead at the bottoms and 2 in. of lead at the sides, and are removable. Electric power is fed to each bridge by 13 slip rings at the center pivot column and to the carriages by seven busbars on each bridge. The slip rings and busbars for each crane are used as follows:

Slip Rings	Bus Bars
1 - common	1 - common
4 - two bridge motors	
2 - lights on bridge	
2 - trolley motor	2 - trolley motor
2 - hoist motor	2 - hoist motor
2 - limit switches	2 - limit switches

The lubrication, brakes, and clutches are the same as in the case of the air-cell crane. The trolleys and hoists of the argon-cell and air-cell cranes are alike.

1.3.10.3 Air Cell Electromechanical Manipulators(13)

The air cell is provided with two 750-lb-capacity, bridge-type manipulators with linear bridges which operate on parallel rails along the sides of the cell. The rails are slightly over 13 ft above the floor, and the manipulators move below the crane and cannot pass each other. The bridges each have four wheels (two on each side) of which one on each side is a drive wheel.

Each carriage which operates on a bridge has four wheels, weighs about 8700 lb, and carries a vertical telescoping arm. The arm can be raised or lowered and rotated. The end of the telescoping arm terminates in a removable working tool. Various tools or mechanisms can be attached to the end of the telescoping arm (forearm). The presently developed tools are:

a) a dual hook grip;
b) a parallel jaw;
c) a general-purpose articulated arm.

The dual hook grip and the parallel jaw are opened and closed by a telescoping rotary linkage to the motor in the carriage above. These mechanisms can thus be opened and closed, rotated, raised, and lowered. They can be moved in the "x" and "y" horizontal directions by carriage and bridge movements. The general-purpose articulated arm has the standard shoulder and elbow-joint movements, shoulder and wrist rotation movements, and a gripper movement, each of which has a separate motor (five in all). When the articulated arm is attached to the manipulator forearm, a plug with eight electric contacts is connected into a socket below the carriage for electric power. This automatically disconnects the circuits to the gripper and arm rotation motors of the telescoping arm and connects these circuits to the gripper and shoulder rotation motors on the articulated arm. The motors on the articulated arm are not shielded, and, when the arm is not attached to the manipulator, it is stored in one of the two 24-in.-diameter storage pits (see Figure 9, page 28) in the floor of the argon cell.

The manipulator motors are all reversible DC motors, the applied voltages ranging from 30 to 100 v. Each motor circuit is fed by a separate isolated rectifier and is arranged for three-speed operation. The motors (except on the articulated arm) are shielded with 4 in. of lead on the bottom and 2 in. on the sides, and are removable.

Electric power is fed to the bridges by 19 busbars on the wall and to the carriages by 16 busbars on the bridge (eight on each side of the carriage). When the carriages are lifted from the bridges, the busbar brushes are automatically lifted from the busbars. The wall busbars are used as follows:

1 - bridge motor
1 - common
1 - small motor field
1 - large motor field
1 - carriage motor
1 - hoist motor
1 - arm rotate motor
1 - gripper motor
3 - limit switches on carriage
1 - gripper force indicator
1 - gripper force control
1 - articulated arm shoulder motor
1 - articulated arm elbow motor
1 - articulated arm wrist motor

This leaves one spare busbar on the bridge and three on the wall. All the wall busbars are connected to the bridge busbars except the one to the bridge motor. The electric wiring on bridges and carriages (except in the motors) is provided with inorganic insulation. Special high-altitude brushes are used to minimize wear in the dry atmosphere.

The bridge and carriage drives have solenoid-operated clutches which fail-open so that the bridge and carriage are free wheeling and can be moved by other equipment in case of drive failure. The arm-rotation drive is provided with both a spring-loaded clutch (enabling the clutch to engage automatically when the motor is running) and a torque-limiting clutch. The hoist motor has a manually (with a manipulator) operated clutch, permitting disengagement prior to removal of hoist drive unit. The grip drive is provided with an electromagnetic clutch which permits remote control of the grip force. The carriages can be lifted from the bridges through the hatch in the air cell ceiling.

1.3.10.4 Argon Cell Electromechanical Manipulators(13)

The argon cell is provided with six 750-lb-capacity, bridge-type manipulators with rotary bridges (see Figure 6, page 25), which pivot on a post at the center of the cell. The bridges operate on circular rails located $11\frac{1}{2}$ ft above the floor at the outer wall of the cell and on circular rails mounted near the center of the cell on the roof of the control area. The manipulators cannot pass each other; however, alternate bridges are raised to permit some overlapping at the inner ends for closer approach to each other. Each bridge has two wheels on the inner rail and two wheels on the outer rail.

Electric power is fed to the bridges by cables which extend down through the central column. The cable to each bridge contains 18 electric circuits. Each circuit consists of three wires or leads in parallel to insure a circuit in case two of the leads should fail. The cables limit bridge movements to about 270° on either side of the neutral position.

The carriages are the same as those described for the air-cell manipulators. They can be lifted from the bridges and lowered to the floor.

Work is being done on the possibility of replacing remotely the manipulator lead-in cables and on using the cranes for certain manipulator repairs.

1.3.10.5 Argon Cell Blister

The crane and manipulator trolleys can be removed from the bridges and lowered to the cell floor by a motor-powered chain hoist in the blister on the cell roof (see Figure 6, page 25). The blister hoist chain is lowered through an 8-in. opening in the argon-cell roof and is guided in the cell by a telescoping tube. The crane or manipulator trolleys are lowered by a special hook attached to the blister hoist when within the cell and can be repaired in the cell or removed from the cell through the large air lock. The 8-in. opening in the argon-cell roof can be closed by a cover with a mercury seal when the chain hoist is in its upper position, and the blister can be opened from the roof for hoist replacement or repair.

1.3.10.6 Crane and Manipulator Controls

The power-control switches for the AC supply to the crane-motor circuits and the DC-supply rectifiers to the manipulator-motor circuits are located in a cabinet in the Operations Control room (see Figure 5, page 24). The manipulator motor-armature circuits have separate transformers and rectifiers to allow control by one lead in addition to the field and common leads. Each motor has three speeds which can be selected from five voltage taps on the power-supply transformer.

Small portable control boxes, each with eight finger-operated (forward and reverse) switches and a grip force indicator. are used to control the cranes, bridge manipulators, and blister hoist. Each switch controls a motion (forward and reverse) including the three-speed steps. Only for a manipulator with articulated arm are all eight switches needed. There are 29 control points at windows where the control boxes can be plugged in. At the power-control cabinet, any control point can be connected to control any crane, manipulator, or the blister hoist. One operator could use, say, two boxes side by side to control two manipulators at the same time. An intercommunication system enables the operators at various locations to communicate with each other.

The performance characteristics of the manipulators are as follows (speeds given are maximum):

Bridge Speed, ft/min	
Rotating Bridge, outer end	13
Linear Bridge	13
Carriage Speed, ft/min	9
Telescoping Arm	
Hoist speed, ft/min	9
Lift capacity, lb	750
Down thrust, lb	150
Arm extension, in.	123
Side thrust (on extended arm), lb	100
Rotational speed, rpm	5
Rotational torque, ft-lb	100
Grip Mechanism	
Parallel jaw hand	
Opening, in.	8
Open-close speed, in./min	27
Grip force, lb	225
Dual hook	
Opening, in.	$2\frac{1}{2}$
Open-close speed, in./min	8
Grip force, lb	750

1.3.10.7 Master-Slave Manipulators

There are six pairs of Model 8 master-slave manipulators in the air cell and two pairs of Model A master-slave manipulators in the argon cell. The Model A manipulators are similar to Model 8, but are provided with gas-tight seals. This was accomplished by using rotating shafts instead of tapes for transmitting movements from the master side of the seals to the slave side of the seals. The inner and outer arms, i.e., the arms on either side of the shielding wall, of the manipulator can be individually removed without breaking the gas seal. The through-the-wall sleeve with seals can also be replaced by shoving the sleeve into the cell with a new sleeve. During sleeve replacement, the integrity of the cell atmosphere can be maintained.

1.3.11 Cell Auxiliaries

The cell auxiliaries consist of rooms and areas with various facilities at the operating and service floor levels in the Fuel Cycle Facility (see Figures 5 and 6, and Table 8).

Table 8

OPERATING LEVEL AUXILIARY SPACE AND EQUIPMENT

Room	Equipment	Purpose
Operating Area	Control Consoles for In-cell Equipment	Main operating area
Mock-up Area	Prototype Electromechanical Manipulator; Spare Model 8 Manipulator; Large Crane (20 ton)	An area in which the feasibility of a proposed remote operation can be determined.
Decontamination Room	Steel-lined Room with Blowout Wall and Sodium Disposal Hood	An area in which sodium can be disposed of and an area where decontamination can be performed.
Degas Room (Measurements Laboratory)	Vacuum-type Induction Furnace; Air Gauges; Optical Comparator; Polariscope; Length-Weight, Diameter, and Porosity Measuring Components of In-cell Equipment; Components of In-cell Welding Equipment.	Melt-refining-crucible preparation; Cladding and Vycor mold inspection; Calibration and test of in-cell equipment.
Mold Preparation Laboratory	Fume Hood; Spray Hood; 2350°F Furnace; 600°F Oven	Mold coating; Crucible coating; Mold baking; Mold baking
Glove Box Laboratory (Dry Box)	Glove boxes (low moisture-low oxygen argon) Vapor Degreasing Tank	Sodium extrusion and loading into fuel element cans; maintenance and service of process equipment.
Machine Shop	Standard Technical Shop Equipment	Assist in the development and maintenance of process equipment.
Operations Control Room	Central Control Panel and Relay Rack for Cell Cranes and Manipulators; Indicator and Control Panels for Stack Gases (radiation monitoring, etc.).	Central control of crane and manipulator operations.

1.3.12 Service Floor Auxiliaries

Subcell (see Figures 3 and 6)

The subcell is located beneath the argon cell at the service-floor level. It also provides entrance to the central control room in the argon cell. Cooling equipment, filters, and purification equipment for the argon cell are located in the subcell. Cell auxiliary equipment, such as melt refining and injection casting vacuum pumps, hold-up tank pumps, silver nitrate iodine tower equipment, and pneumatic and electrical equipment which might be contaminated, are located here. The floor feedthrough sleeves for equipment in the argon cell terminate in the subcell.

The following auxiliary equipment is located in the service-floor areas:

1) argon-supply system including storage cylinders;
2) refrigeration condensing units for cooling of cell atmosphere;
3) induction-heating motor-generator equipment;
4) argon-recycle system for cooling the injection casting furnace.

The following service equipment is located in the service-floor area:

1) ventilation fan and filter system;
2) heating system;
3) water, steam, drainage, and service systems.

1.3.13 Sodium-disposal System

The system for disposal of residual coolant sodium within a subassembly consists of a venturi scrubber and separator, a suspect retention tank, an air heater and filter, and a turbo-compressor, all located on the service floor. The compressor draws air from the 10- by 10- by 8-ft high sodium-disposal box in the decontamination room and from the 4-in. sodium wash water drain in the interbuilding passageway. The air is drawn through a venturi scrubber to remove any sodium and activity which may be present, through an entrainment separator, and then through a heater and an AEC filter. The air then passes through the compressor and is discharged to the suspect stack through the building ventilating system. The water from the scrubber is discharged to the retention tank. The compressor, which is rated at about 3500 cfm at 38 in. of water suction pressure, is located on the south side near the building ventilating fans; the remainder of the equipment is located in the southwest corner of the service floor. The retention tank is 6 ft in diameter and 8 ft long and holds about 1700 gal.

1.3.14 Analytical Cave Facilities

There are six analytical caves in the Laboratory and Services Building. The caves are in a row, and each cave is 6 ft wide by $4\frac{1}{2}$ ft deep. The caves are interconnected by a 2-ft-high by 1-ft-wide conveyor passageway. Each cave has an access door opening to the room and each has a 56-in.-wide by 40-in.-high shielded window and a pair of master-slave manipulators. The walls of the cave are composed of 220-lb/ft^3 concrete and are 2 ft thick.

The gamma-attenuation factors for the 2-ft-thick walls are as follows:

Gamma Energy (Mev)	Thickness for Attenuation Factor of 10, No Buildup (in)	Buildup Factor	Total Attenuation
2.5	6.5	9.2	550
1.7	5.4	18.4	1500
0.8	3.7	120	25000

The gamma-attenuation factors for the 2-ft-thick windows are as follows:

Gamma Energy (Mev)	Attenuation Coefficient Times Thickness	Buildup Factor	Total Attenuation
2.5	7.82	6.8	370
1.6	9.03	7.2	1150
0.75	15.40	8.8	546000

The activity per gram of the fuel at 2 a/o burnup in 135 days is as follows:

	Curies of Gamma of Stated Energies		
Days Cooling	2.5 Mev	1.6 Mev	0.8 Mev and Less
15	0.185	3.7	40.6
30	0.081	1.64	33.3
60	0.018	0.36	22.6
120	-	-	13.1
240	-	-	5.7

The gamma radiation in mr/hr at the outside surface of the windows and walls due to one gram of fuel (having 2 a/o burnup in 135 days) positioned 1 m from the inside cave wall would be as follows:

	Gamma Radiation (mr/hr)	
Days Cooling	Window	Wall
15	3.18	3.10
30	1.43	1.65
60	0.32	0.64
120	0.01	0.23
240	0.005	0.10

In order to handle $1\frac{1}{4}$-g samples of irradiated fuel pins and similarly sized samples of skulls or skull oxides (which would have activities about five times as great as the fuel pins), additional shielding inside the caves (such as lead bricks) would probably be desirable.

2. FUEL-PROCESSING OPERATIONS AND ASSOCIATED CONSIDERATIONS

2.1 Fuel-processing Operations and Equipment

The objectives of the fuel-processing operations are the recovery and purification of discharged fuel material and refabrication of the recovered fuel into new fuel elements. The major steps in the processing cycle were previously presented in Figures 1 and 2. These major steps are disassembly of the discharged fuel, purification by melt refining, refabrication of new fuel pins, and reassembly of reactor core subassemblies.

The fuel-handling and -processing equipment, which is being initially installed in the Fuel Cycle Facility, is designed for processing the first type of fuel loading of the reactor core (uranium-5 w/o fissium alloy as described in Table 1). The flowsheet for this process has been described by Ader.(14) Initially, the skulls or skull oxide from the melt refining operation will not be processed, although equipment for this purpose is being developed for later installation into the Fuel Cycle Facility. The recovery of plutonium from blanket material is part of the complete process. Demonstration equipment is being developed for the two latter processes (skull reclamation and blanket). These two processes are described briefly in Section 2.2. Much of the fuel-processing equipment may be suitable for subsequent types of fuel loadings.

The location of the initial equipment in the argon cell and air cell is shown in Figure 11. The general layout of the subcell and location of major equipment items in the subcell is shown in Figure 12.

2.1.1 Fuel Transfer via Interbuilding Subassembly Transfer Coffin

Fuel will be transferred from the reactor to the Fuel Cycle Facility in a heavily shielded transfer coffin which will hold one subassembly. The primary purpose of the subassembly transfer coffin is to provide radiation shielding for reactor subassemblies during transfer. Auxiliary purposes are to provide cooling to remove the heat generated by the fission products and to isolate from gross quantities of air the sodium adhering to the subassembly until the sodium can be oxidized and washed off by water.

The coffin will be a top-loading device with a sealed plug closure and with an interior chamber about 3 in. in diameter and 95 in. long. The coffin is designed to be moved and handled in a vertical position at all times. The subassembly is oriented in the coffin by a bar which fits the slot in the bottom adapter.

The total self-heating effect of a fuel subassembly containing 6070 g of fuel with 2-a/o burnup in 135 days and 15-day cooling is about 1220 w if all of the beta and gamma energies are absorbed, as shown in

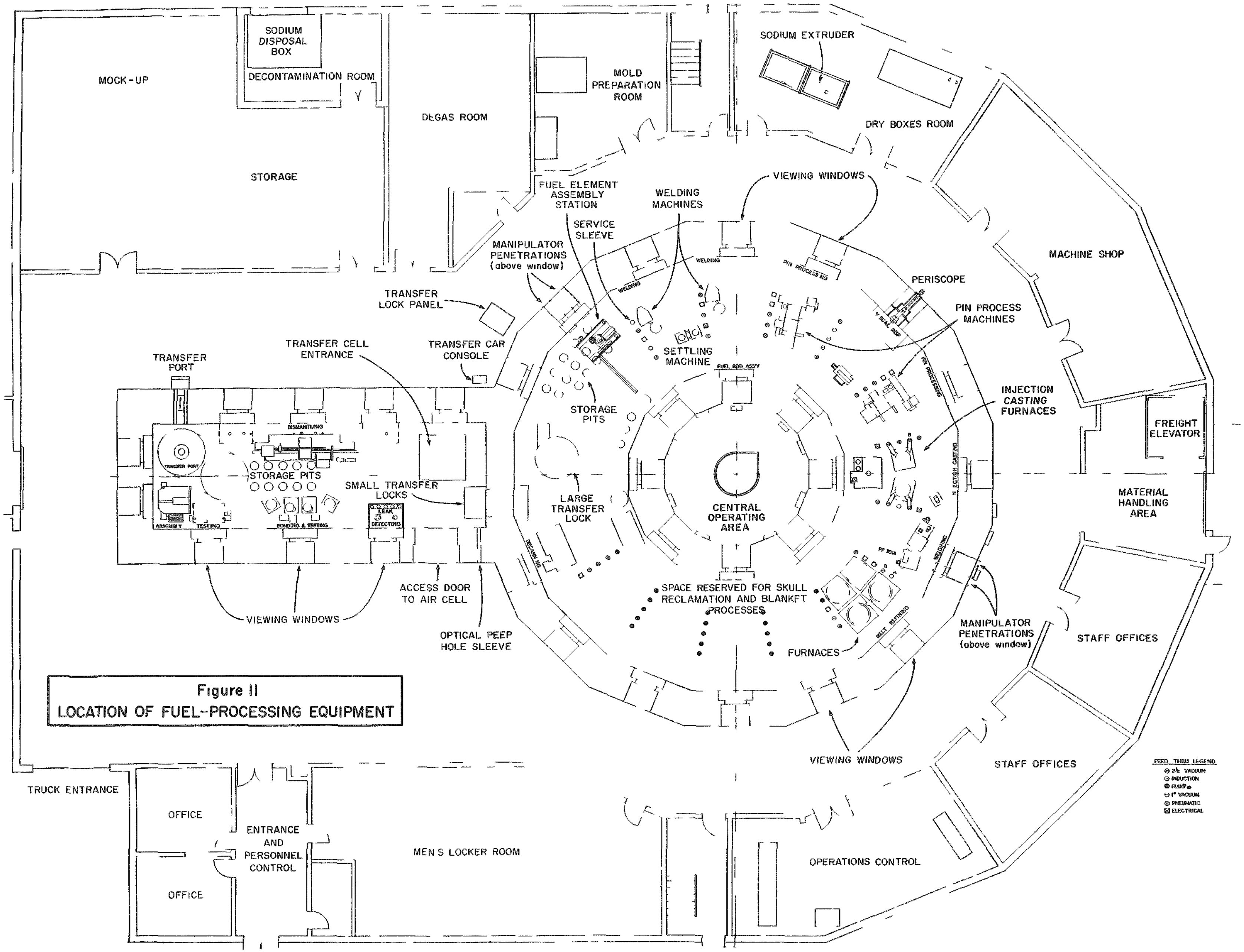


Figure II
LOCATION OF FUEL-PROCESSING EQUIPMENT

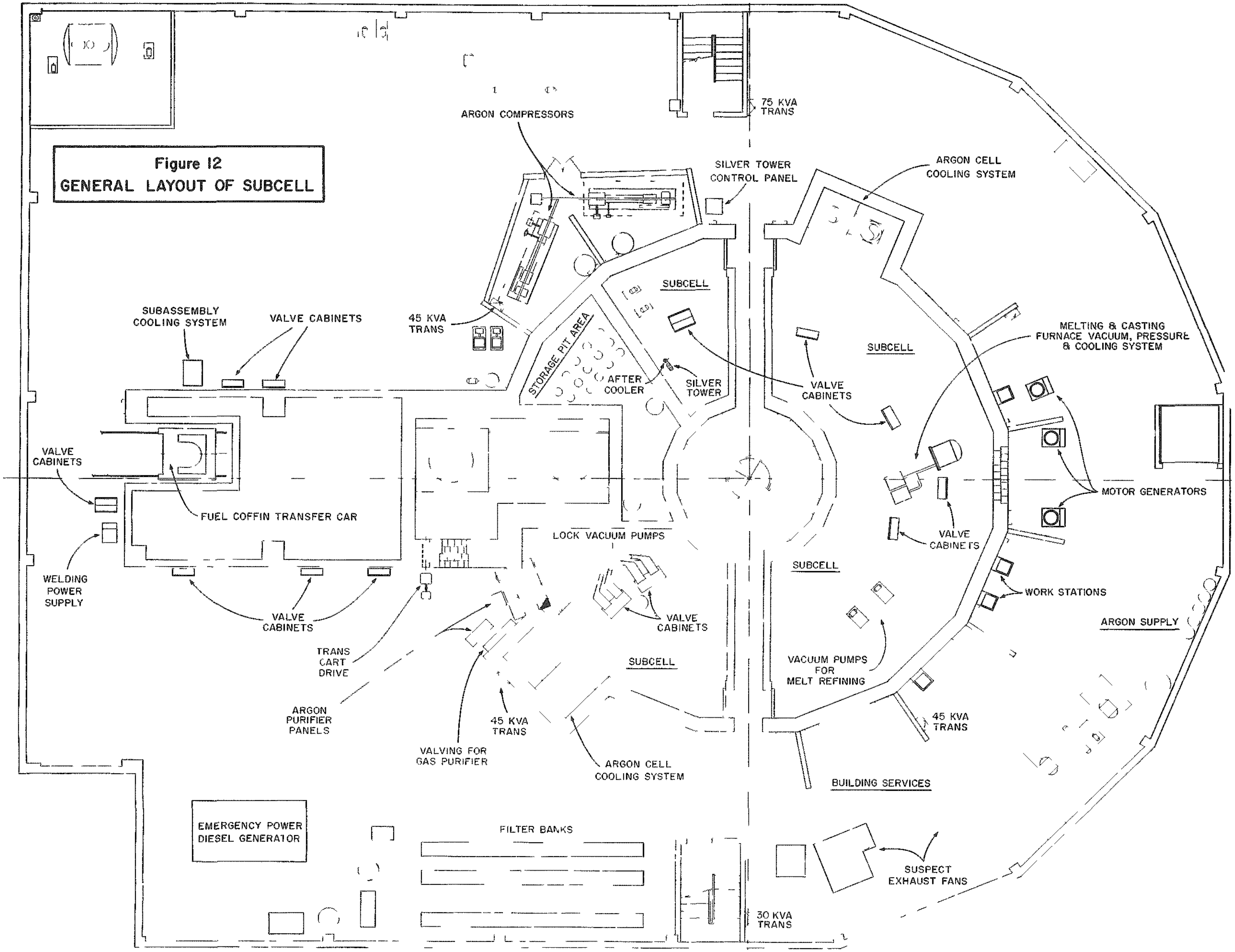


Figure 12
GENERAL LAYOUT OF SUBCELL

Table 5. The energy of the gamma radiation averages about 0.75 Mev. Of the total energy, about 2 percent is 2.5 Mev, 25 percent is 1.6 Mev, and the remainder is less than 0.8 Mev. The attenuation coefficients for these values for lead are about 0.87, 1.00, and 1.78 per centimeter, respectively. The buildup factor for 28 cm of lead (which is the approximate thickness of shielding on the coffin) for these values is about 15. In order to reduce the radiation level to less than 0.1 mr/hr at the surface of the coffin next to the center of the fuel subassembly, about 11 in. of lead would be required. The shielding will consist of lead which will be about 13 in. thick on the sides at the fuel level, 10 in. thick on the top and bottom, and 7 in. thick on the sides below the fuel level. The interior and exterior of the coffin will be lined and covered with stainless steel and sheet steel, respectively, at least $\frac{1}{4}$ in. thick. The coffin will weigh about 15 tons.

Interbuilding Subassembly Transfer Coffin

Two types of coffins are being constructed. These coffins differ mainly in their provisions for cooling the subassembly. In the first type, which is being procured from an outside manufacturer, cooling is effected by an external blower system which circulates the coffin atmosphere from the bottom of the coffin through an air-cooled heat exhcnager, through a blower, and then down through the subassembly in the coffin chamber. The heat exchanger is provided with a fan for circulating the cooling air.

In the second type, which is being manufactured at the Illinois site of Argonne National Laboratory, the cooling is accomplished by transferring the heat from the subassembly to the lead shielding. The interior stainless steel chamber is provided with heat transfer fins to transfer heat from the circulating atmosphere to the lead shielding. An external blower system circulates the coffin atmosphere from the top of the coffin, possibly through a filter, through a blower, down through the subassembly in the coffin, then up through the heat transfer fins surrounding the subassembly for heat removal, and then out of the top of the coffin to the filter.

The coffins will be provided with a supply tank of argon for the circulating atmosphere.

2.1.2 Sodium Removal from Subassembly

The purpose of the operation of sodium removal is to remove sodium which adheres to the subassembly while the subassembly is in the passageway between the Reactor building and the Fuel Cycle Facility. The sodium to be removed is from the reactor coolant system and has short half-life activity. The half-life for sodium-24 is 15 hr, and the activity on removal from the reactor will be about 1.5 mc/g of sodium. It is desirable to remove the sodium while the subassembly is in the relatively inert atmosphere of the coffin. The pressure of the gas in the coffin chambers will be atmospheric or perhaps slightly above. If air were admitted through a slight leak in the coffin sealing gasket, the sodium would merely oxidize slowly.

At 400°C, the density, viscosity, and surface tension of sodium are 0.86 g/cm^3, 0.285 centipoise, and 160 dynes/cm, respectively; in comparison, values for water at 20°C are 1.0 g/cm^3, 1.0 centipoise, and 75 dynes/cm, respectively. From these data, one would expect sodium retention at 400°C due to capillary action to be greater than would be the case for water at 20°C.

The design of the system for sodium removal has not been decided; however, the present plan is to remove the sodium in steps as follows:

a) Admit oxygen diluted by nitrogen to the circulating gas system to oxidize the sodium.

b) Admit water vapor in nitrogen to allow further reaction with the sodium.

c) Flow water through the coffin and subassembly to remove the oxidized sodium and assure that all sodium is oxidized.

d) Immediately after the water wash, blow out the water and dry the coffin subassembly with a stream of air. In case of need, a preheater for the air would be available.

The wash water and vent gases are conducted to a 30-gal hold-up tank through a 4-in. stainless steel pipe (the sodium wash water drain pipe). Air, which is constantly drawn through this pipe by the building exhaust system at a rate of about 200 cfm, serves to dilute any hydrogen gas and to minimize the release of contaminated vapors. A gas-water separation is made at the hold-up tank. The 4-in. pipe continues on to the intake of a venturi scrubber. The air is scrubbed in the venturi scrubber to remove any sodium and activity which may be present, passed through an entrainment separator, and filtered before being discharged up the suspect vent stack.

The hold-up tank will provide for monitoring of wash water which will be contaminated with active sodium and, in the case of failure of a fuel-element cladding, with fission products. A pump for pumping the contents to the sodium-disposal-system hold-up tank on the service floor will be incorporated.

2.1.3 Operation of Interbuilding Subassembly Transfer Coffin

The sequence of transfer operations in which the coffin is always in a vertical position starting with the coffin in the reactor passageway is shown in Figure 13.

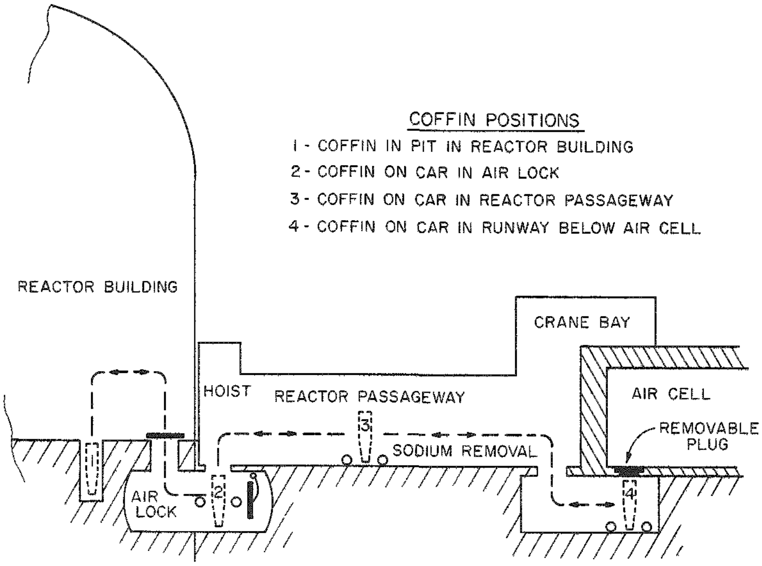


Figure 13

MOVEMENTS OF SUBASSEMBLY COFFIN BETWEEN REACTOR AND FUEL CYCLE FACILITY

The subassemblies are placed into and removed from the coffin in position 1 in the reactor building by the refueling machine. The coffin is loaded onto and off the carts in the air lock, the reactor passageway, and under the air cell by the cranes and hoist in the reactor building, reactor passageway, and crane bay, respectively. The subassemblies are placed into and removed from the coffin in position 4 under the air cell by the crane in the air cell.

There are provisions for supplying electric power (normal and emergency power) to operate the coffin cooling blower throughout the cycle of coffin movement. The coffin will be supplied with temperature and radiation alarms. It will be dried and purged with argon before return to the reactor building.

2.1.4 Dismantling Subassemblies

The purpose of this operation is to dismantle the subassemblies to be processed so that the fuel elements (canned fuel pins) can be recovered as individual items for transfer to the argon cell for processing and so that the blanket elements (canned blanket pins) can be recovered as individual items for disposal, storage, or transfer to a plutonium-recovery facility.(15)

There will be 91 fuel elements and 36 blanket elements (20 in. long) from each core subassembly, 61 fuel elements and no blanket elements from control and safety subassemblies, and 19 blanket elements (62 in. long) from each blanket subassembly. The inner blanket pins will be processed after approximately one percent buildup of plutonium. (This would require an estimated time of 2.6 yr of constant reactor power of 62.5 Mw for the blanket elements next to the core. The blanket uranium pieces in canned fuel pins would have a very low plutonium content after 135 days of irradiation.) When the fuel reaches 2-a/o burnup, the upper and lower blanket pins of the core subassemblies would have only about 0.14 percent buildup of plutonium at the ends next to the core and essentially no plutonium buildup at the other ends. Although, initially, new blanket elements will be reassembled with the processed core subassemblies, eventual consideration might be given to reassembling the irradiated end blanket elements in processed core subassemblies to increase the plutonium buildup in these pins. If the end-blanket elements were to be reassembled in processed core subassemblies, greater care in disassembly would be required to avoid damaging the grids, end pieces, and blanket elements.

2.1.4.1 Cooling of Subassemblies

During handling and disassembly of the core subassemblies, cooling will be required to remove the heat generated by the fission products.(16-18) Just before the subassembly is lifted from the coffin and into the air cell, a flexible air hose is connected by the grapple to the upper end of the subassembly. Air can be forced or drawn through the subassembly for cooling while it is being lifted into the air cell and moved to the disassembly station. At the disassembly station, another flexible air hose with provisions for a pressure of about 10 psi at 25 to 40 ft^3/min (100 to 160 lb/hr) is connected to the lower end of the subassembly. When the hexagonal tubing is pulled off the fuel elements, a flow of air through the fuel elements from a cooling plenum (see Figure 14) is initiated to cool the fuel elements. The use of suction to cool the elements is being considered to avoid possible spread of contamination.

Self-heating is the result of absorption of about 65 percent of the gamma and essentially all of the beta energies in the subassembly. For 2-a/o burnup in 135 days and 15-day cooling, the heat absorbed would amount to about 1000 w per core subassembly. In order to cool the subassembly and keep the fuel at a temperature low enough to provide a factor of safety, about 100 lb of air at about 85°F passed through it should be adequate. This would give an air speed of about 40 ft/sec along the fuel elements. The temperature rise of the air would be about 180°F (100°C), and the average temperature difference between the fuel elements and the air would be about 90°F (50°C). The pressure drop through the subassembly

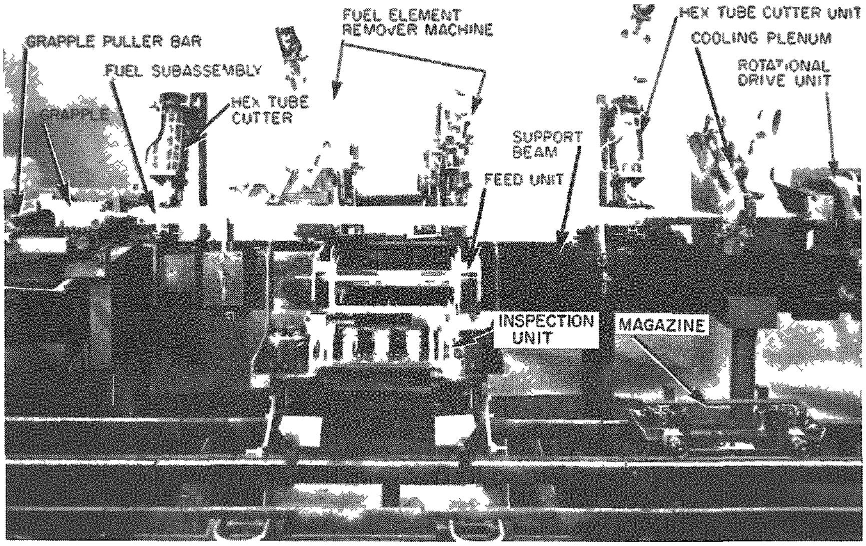


Figure 14

EBR-II SUBASSEMBLY DISMANTLING MACHINE(15)

would be of the order of 2 psi. For this rate of airflow, a pressure system might be preferable to a suction system. For the case in which the subassembly is lifted from a vertical coffin, a suction system would be desirable from the viewpoint of contamination containment. After the subassembly was removed from the coffin, a pressure system could be attached.

The specific heat of uranium is about 0.03 cal/(gm)(°C) and for stainless steel about 0.13 cal/(gm)(°C). If the air cooling were stopped, the temperature of the fuel elements would rise, initially, at the rate of about 1.8°F (1.0°C) per second. Hence, if cooling were lost for as much as 15 min, the center of the assembly might overheat. It is estimated that the center fuel element would attain a maximum temperature of 1590°F (870°C)(18) (for a subassembly in the open air). This temperature might cause alloying between the uranium fuel and stainless steel, or swelling and possible rupture of the element cans. These possible effects do not seem serious enough to warrant a third (emergency) cooling system, such as a liquid bath, to supplement the available suction or blower systems.

2.1.4.2 Cooling of Fuel Elements

If the disassembled fuel elements are placed in a single layer for storage or movement, essentially all of the beta and about twenty percent of the gamma energies would be absorbed. The heat per unit of 91 elements would be about 600 w. The top and bottom surface area of the layer of pins available for heat radiation will be about four ft^2. The worst condition from a cooling viewpoint would occur while the elements are under vacuum in the transfer lock to the argon cell, where heat transfer would be by radiation only. For an emissivity of 0.5, the elements would reach an equilibrium temperature of about 510°F (267°C) at an ambient temperature of 100°F. If only one side of the element layer were exposed for radiation, the equilibrium temperature would be about 665°F (350°C). In the open, or in the argon or air cell, convection cooling would also occur and the equilibrium temperature would be less. It appears that radiation cooling (and cooling by natural convection) is sufficient for fuel elements if they are in single layers and free to radiate from at least one side to a background at normal room temperature (100°F).

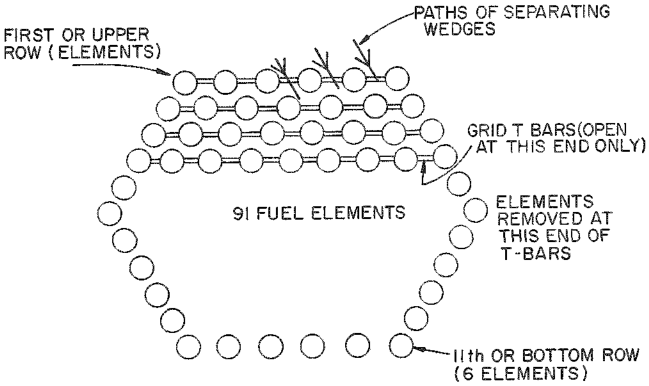
2.1.4.3 Disassembly Machinery (15)

During disassembly, the subassembly is held in a horizontal position. The disassembly apparatus somewhat resembles a lathe (see Figure 14). For the circumferential cutting operations, the subassembly is rotated by an electric-motor drive which engages the slot in the bottom adapter of the subassembly. The subassembly is held in position by a split bearing which fits around the lower adapter between the coolant slots and the hexagonal (hex) tubing. The hex tubing rests in two spaced cradles. A rotary plenum for cooling air is clamped around the coolant slots. This plenum also acts as a thrust restrainer. The apparatus is provided with three spring-loaded, pipe-type, pneumatically operated roll cutters for cutting the hex tubing at each end and for cutting the bottom adapter as the assembly is rotated. One cutter is located near each end of the tubing and one at the bottom adapter. The cutters swing upward and backward when not in use. The end of the apparatus supporting the top end of the subassembly is provided with a chain-operated, axial pulling device which has a fork for engaging the end piece and jaws for seizing the hex tubing after the end piece is removed. This device resembles a small draw bench and moves axially to pull the top piece with upper grid and blanket elements out of the hex tubing and to pull the hex tubing off the bottom adapter after the hex tubing has been cut at the top piece and bottom adapter, respectively.

The disassembly apparatus also includes a powered socket wrench for twisting off the hex nuts on the grid spacer rods. It is also equipped with two roll cutters which move in the axial direction of the subassembly. These cutters are used to slit the hex tubing on two opposite sides so that the hex tube can be split in two pieces. This cutter would be used for blanket and possibly control subassemblies, and in cases in which difficulty is encountered in pulling the elements from the tubing (such as difficulty caused by fuel swelling).

There are four grids in the core subassemblies for holding the blanket and fuel elements in axial position. Two of the grids are attached to the top piece and bottom adapter of the subassembly. The other two grids are attached to the bottom and top grids by center spacer rods which have nuts on their lower ends. The blanket elements are held between a pair of grids. The fuel elements are held by a grid at their lower ends only. The lower end plug of each blanket element is welded to its grid, and the upper end plug of each blanket element floats in its grid.

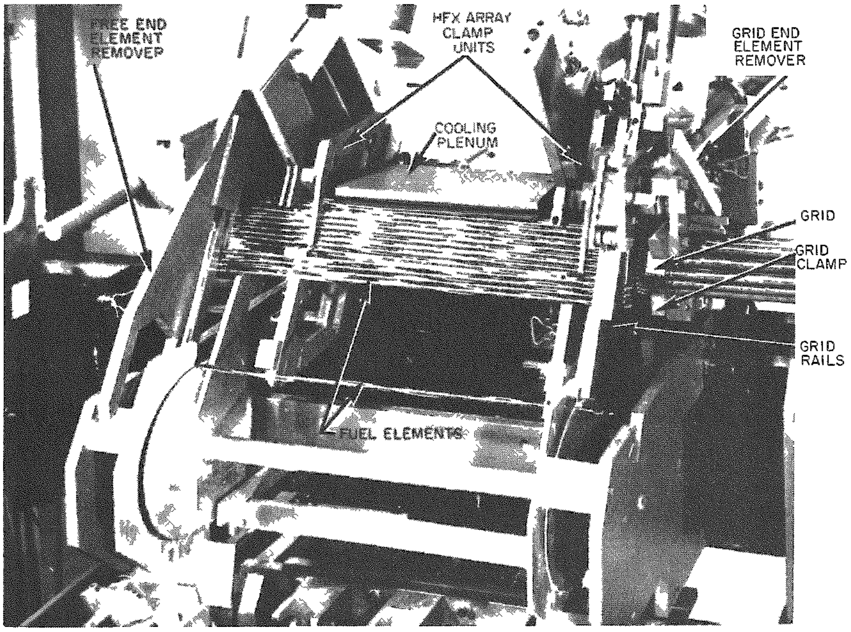
In the core section grid there are eleven parallel T-bars. The first and eleventh bars hold six fuel elements each, the second and tenth bars hold seven fuel elements each, and so on. The sixth or center bar holds eleven elements (see diagram).



A hexagonal array clamping device with separating wedges is used to separate and remove the fuel elements from the T-bars. The air plenum for cooling the fuel elements during removal is attached to the clamping device (see Figure 15). The subassembly is rotated to a position where the grid T-bars are horizontal and open outwards. The hex tubing and bottom adapter are marked to indicate the open ends of the T-bars.

Figure 15

FUEL ELEMENT REMOVING MACHINE (FERM)[15]



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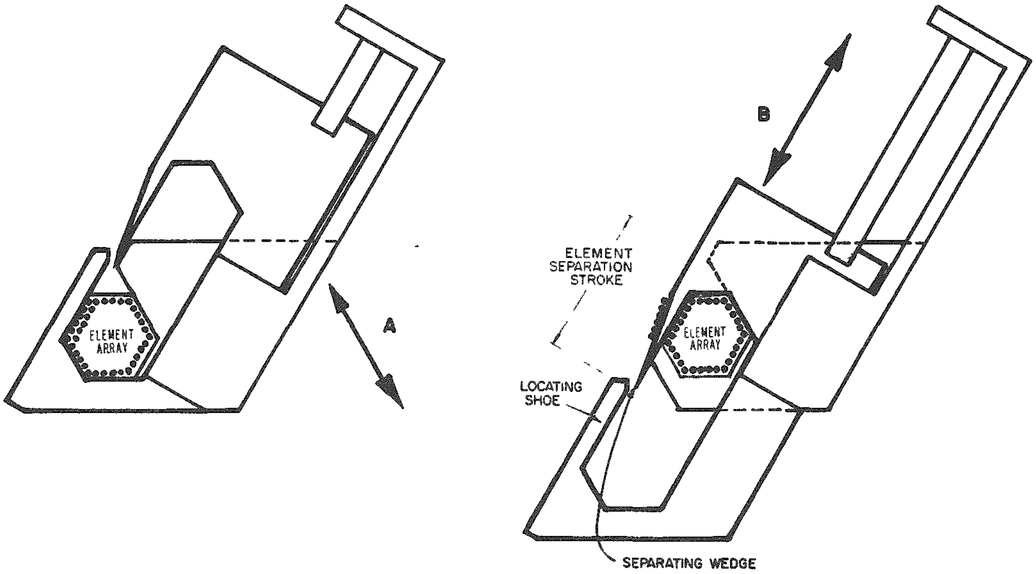
The separating wedges of the hex array clamping device move downward and outward (at 30°) to remove a row consisting of one fuel element from each T-bar in their path per operation (see Figure 16). Thus, on the first operation a row of six fuel elements is removed, i.e., one element from each of the six upper bars. On the second operation, a row of seven elements is removed (i.e., one element from each of the seven upper bars), and so on until, on the eleventh and last operation, a row of six fuel elements is removed (i.e., one element from each of the

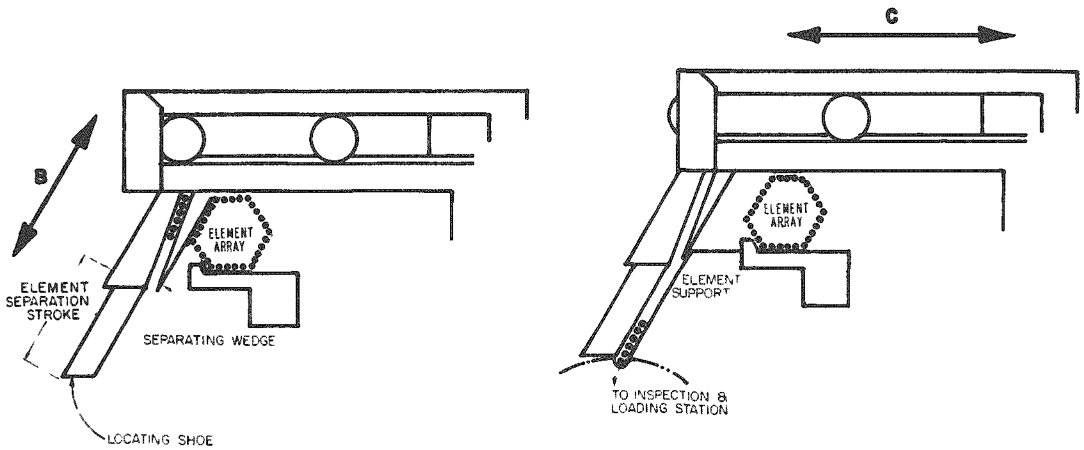
Figure 16

OPERATIONS OF THE FUEL ELEMENT REMOVAL MACHINE(15)



A HEX ARRAY CLAMP
FIRST ROW READY POSITION

B HEX ARRAY CLAMP
FIRST ROW SEPARATED POSITION



C GRID END REMOVER
FIRST ROW SEPARATED POSITION

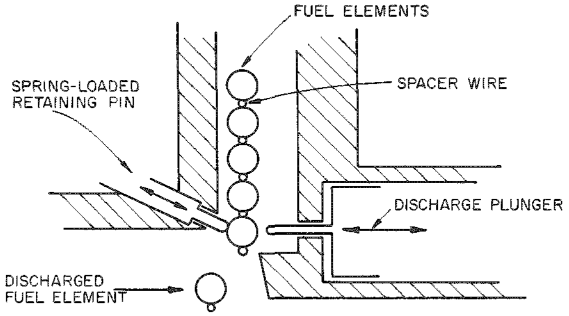
D GRID END REMOVER
FIRST ROW DISCHARGE POSITION

six lower bars). As each row of fuel elements is removed from the T-bars, it drops downward to a rotary feed mechanism which removes the elements in order and passes them to the inspection device which checks them for straightness.

In the inspection device, the fuel element is held horizontally, and a series of movable electric contacts approach the element. The travel of the contacts is limited to a predetermined distance from the axis of the fuel element. If contact is not made with the element, it is accepted and is fed into the top of a pin-carrying magazine. If electric contact is made, the element is not straight enough for regular decanning and is removed from the machine for special handling by means of a master-slave manipulator. Because the contacts form a 1-in.-diameter cylinder concentric with the element axis, the effective criterion for straightness of the fuel element is that it should pass through a 1-in.-inside-diameter tube. The position which each fuel element occupied in the subassembly can be identified by a record of the position in the magazine in which the fuel element is placed.

2.1.4.4 Fuel-element Magazines and Carts

The magazines each hold from 18 to 25 fuel elements horizontally positioned in a single vertical row. The elements are held by U-slots at each end of the magazine. They are normally retained by spring-loaded pins at the bottom and by spring-loaded levers at the top. The elements are dropped into the magazine one at a time (with the retaining levers withdrawn in the loading brackets) and can be removed one at a time at the bottom by inserting discharge plungers at each end (see sketch below). At the disassembly apparatus, the carts with magazines are set on tracks below the element inspection device (see Figure 14).



The magazines are about 21 in. long, $4\frac{1}{2}$ in. wide, and $6\frac{1}{2}$ in. high. They are locked to four-wheel carts which fit the track in the small transfer locks. Overall dimensions of a cart with magazine are $12\frac{1}{4}$ in. wide, 24 in. long, and $6\frac{1}{2}$ in. high. The carts and magazines are provided with handles to allow handling with a manipulator. Normally, the magazines are removed from the carts only inside the argon cell. There the magazine is used to feed fuel elements to the decanning apparatus.

2.1.5 Fuel-element Decanning and Pin Chopping(19,20)

The purpose of the decanning operation is to remove the stainless steel cladding can from the fuel pins. In separating the stainless steel cans from the fuel pins, it is desirable that very little of the fuel (preferably less than 0.1 percent) be lost with the stainless steel scrap. It is also desirable that very little of the stainless steel cladding be included with the fuel pins. There are many possible sources of cross contamination. The fuel pins may become brittle or friable during burnup and may tend to break during decanning operations. In some cases the fuel elements may be warped or bent to some extent, or the cladding might be ruptured or embrittled. The fuel might be alloyed to the cladding in spots.

It is planned to sample the cladding scrap to determine fuel loss. There are about 9 g of 9-mil cladding per fuel pin, and the estimated activity is 3 c/g of cladding. Since fuel losses might include pieces of fuel pins not attached to the cladding as well as pieces attached to the cladding, fairly large samples of highly active material may be required.

Two types of mechanical decanning mechanisms have been experimentally tried.(19,20) In both types, the fuel elements are first put through a double shear which shears off both ends of the can. The distance between shears is about 16 in. At the lower or spade end, the element is sheared at the fuel-pin end. At the upper end, the element is sheared about $1\frac{3}{4}$ in. above the fuel-pin end. This releases the spacer wire and leaves the fuel pin with the 9-mil stainless steel cladding, part of the top spacer, and the bonding sodium. The shears are indexed as to distance from the spade end. A special arrangement is under consideration for fuel elements having missing or broken spades.

In the first type of decanner, called the spiral decanner,(19) the sheared element is pushed endwise through a die (to make sure that the spacer wire is removed) and then into the center of a nest of three short, knurled rollers which are power driven (see Figures 17, 18, and 19). The rollers are $\frac{3}{4}$ in. in diameter by $\frac{1}{2}$ in. in overall length, with a $\frac{3}{16}$-in. tapered section. They have their axes at an angle of 15° to that of the pin.

In the second type of decanner, called the roll decanner, the sheared pin with cladding is pushed endwise into two roll cutters which are power driven and which have their axes parallel to each other and at right angles to the pin axis. The roll cutters cut the cladding on opposite sides of the pins. In this operation, the fuel acts as an anvil for the roll cutters. The cladding comes off the pin in two pieces. The fuel pin is chopped and samples taken, as in the case of the spiral decanner.

Figure 17

EBR-II FUEL DECANNING OPERATIONS

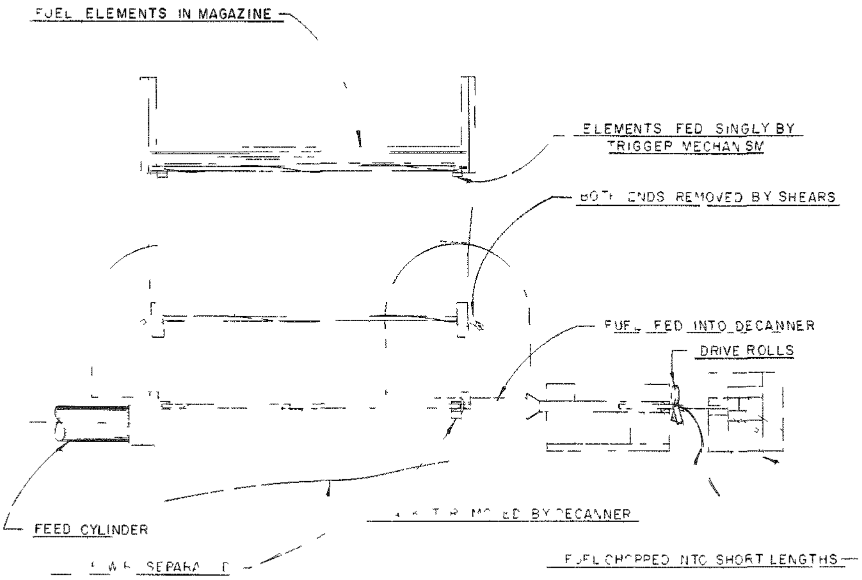


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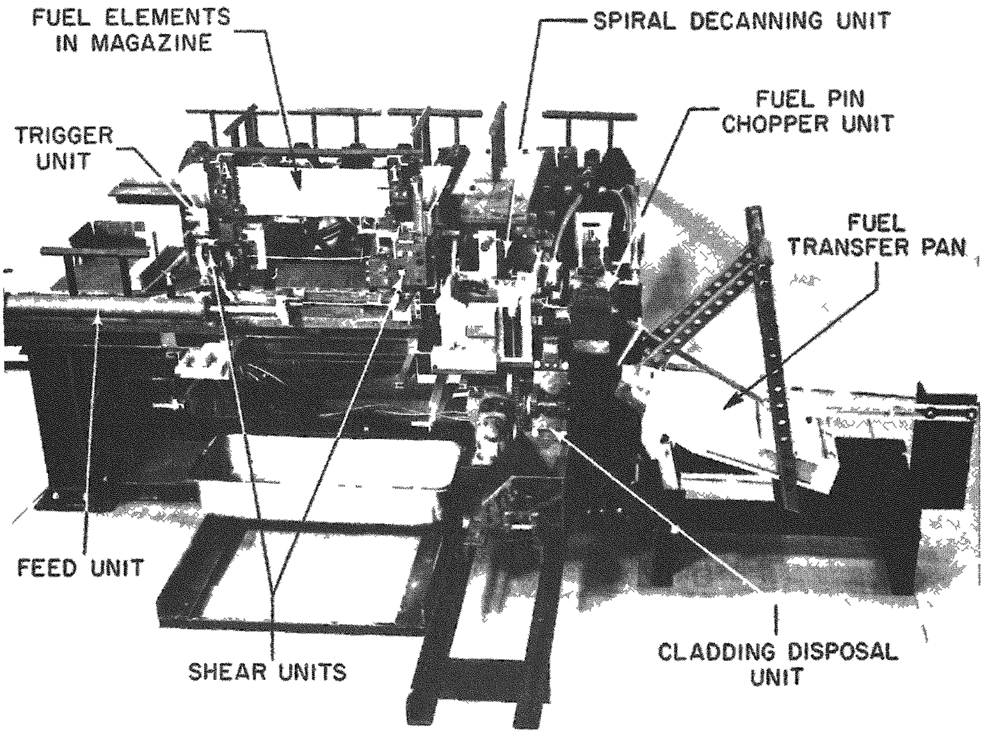
Figure 18

EBR-II FUEL DECANNING MACHINE



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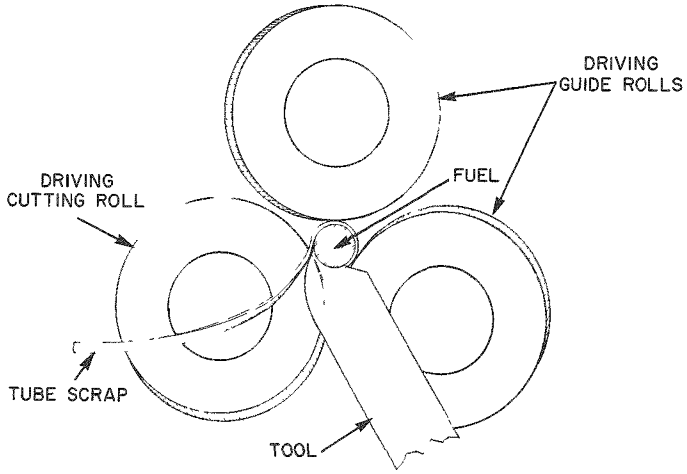

Figure 19

CLOSEUP OF SPIRAL DECANNING OPERATION

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The spiral decanner has been selected for the operation; however, a roll decanner may be added later as a special item for decanning crooked or ruptured fuel elements, since the spiral decanner will not handle badly bowed fuel elements or those with ruptured cans.

After decanning, it is necessary to cut the fuel pins to about $1\frac{1}{2}$-in. lengths to facilitate charging to the melt refining operation.

2.1.6 Melt Refining(3,4,21-36a)

The primary purpose of melt refining is to remove the fission products from the spent fuel. In addition, the melt refining step provides the most suitable point at which fresh fuel materials (to compensate for burnup losses) can be added. In a later stage of the operation it will also serve as a point to return the product of the "dragout" or Skull Reclamation Process to the cycle.

2.1.6.1 Process

The melt refining process consists of charging the decanned fuel from the reactor plus any recycled fuel into a zirconia crucible and liquating at 1300 to 1400°C for about one to three hours under a high-purity argon atmosphere. The volatile fission elements are removed by volatilization. The reactive fission elements are removed by oxidation with the zirconia to form a reaction layer on the wetted crucible surface, which remains in the crucible after the fuel metal is recovered by pouring into a cold graphite mold (a copper mold has also been tried). Some uranium metal is also oxidized and, together with several times as much metallic uranium or fuel, remains with the reaction layer to form a skull in the crucible. The ratio of wetted area of the crucible to weight of charge will be approximately 0.03 cm^2/g. Dry-pressed five percent lime-stabilized crucibles have been found to have better thermal shock resistance than pure zirconia crucibles.

2.1.6.2 Fission Product Behavior

Since high decontamination is not achieved by melt refining, the fuel must be refined and fabricated in shielded cells. The fission products fall into three groups which are important from a melt refining viewpoint, namely, (a) those which volatilize (volatile elements), (b) those which oxidize to form a dross which remains in the crucible as part of the skull after pouring (reactive elements), and (c) those which remain with the fuel and are not selectively removed (noble elements).(23,25) For some time after reactor startup, the fuel lost by burnup and in the melt refining skull will be replaced by fresh fuel material. Under this condition, the noble fission elements would build up with each successive reactor pass until an equilibrium would be reached wherein the amount produced per reactor pass would be equal to that removed with fuel in the metal part of the skull. This removal is known as "dragout." Since only about 5 to 10 percent of the fuel is retained in the skulls, this group of fission elements would build up to an equilibrium which is five to ten times as great as that produced per reactor pass.

Ultimately, the fuel contained in the skulls will be recovered in a skull recovery operation (described in a later section of this report). In this process, removal of noble elements will not be expected to be complete. Thus, their equilibrium level in the fuel will be somewhat above that for direct fuel replacement.

Volatile and Gaseous Fission Products

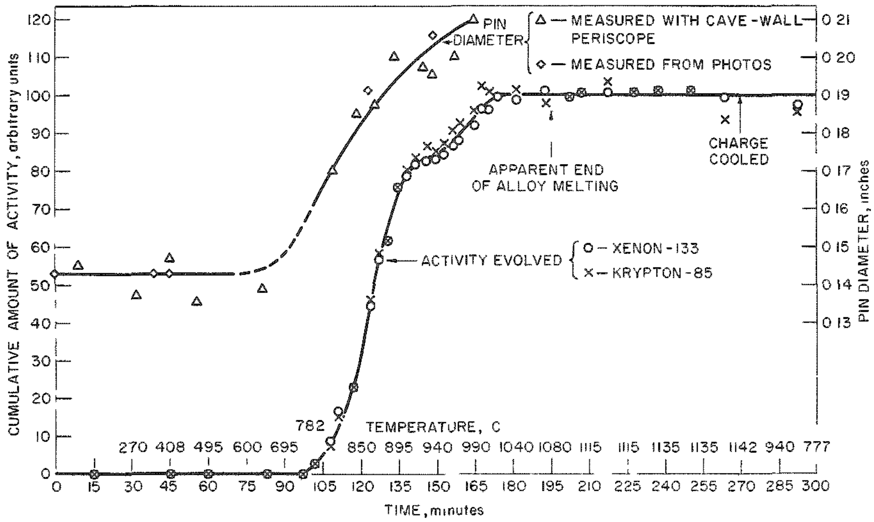
The following fission product elements constitute the first or volatile group; it has been shown that nearly 100 percent removal of these elements will occur in melt refining.(29,30,33-35, 37a)

Element	Symbol	Atomic Number	Weight Percent Formed per Reactor Pass at 2-a/o Burnup
Bromine	Br	35	0.0013
Krypton	Kr	36	0.023
Rubidium	Rb	37	0.029
Cadmium	Cd	48	0.0023
Iodine	I	53	0.012
Xenon	Xe	54	0.260
Cesium	Cs	55	0.23

The pattern of release of the noble gases xenon and krypton was first studied by Schneider and Chellew(29) with lightly irradiated material and more recently by Chellew and Steunenberg(37a, 38, 39) with highly irradiated material. The latter studies, which are considered the more pertinent, were conducted with EBR-II fuel pins which had been irradiated to a burnup of about 0.6 percent in the MTR reactor. Figure 20 shows the change in pin diameter and the pattern of xenon and krypton release on heating a pin to its melting point (about 1000°C). Up to nearly 750°C, little change in pin diameter and little evolution of xenon and krypton occurred. Both swelling and gas release became rapid at about 750°C and were complete by the time the melting point was reached. The ultimate increase in pin diameter was about 50 percent. In another experiment the heating rate was doubled, but both swelling and fission gas release showed the same temperature dependence, indicating that temperature rather than time has a predominant effect.

Figure 20

EXPANSION AND FISSION GAS RELEASE DURING HEATING OF HIGHLY IRRADIATED EBR-II FUEL PIN



ALLOY COMPOSITION, URANIUM-5 WEIGHT PERCENT FISSIUM, LENGTH, 1 5 in; DIAMETER, 0 143 in, TOTAL ATOM BURNUP, ~0 6 PERCENT; COOLING TIME, 49 DAYS
ATMOSPHERE HIGH PURITY ARGON (closed system); PRESSURE, 685 - 790 mm

Although considerable swelling of the fuel pins occurs before melting, it does not appear that cracking of the zirconia crucible will occur, since the fuel-alloy pins have little mechanical strength in the temperature region in which swelling occurs. Neither does it appear that the release of fission product gases during melting will cause spattering of the metal, since the fission gases escape gradually through microcracks and voids during the heating period preceding melting.

The xenon and krypton will be pumped from the melt refining furnace to a gas-holdup tank (described in Section I of this report) to await favorable meteorological conditions before being released through the suspect vent stack. During holdup of the gas, considerable decay of the 5.3-day xenon activity will occur.

Iodine is also essentially completely removed in melt refining.(33, 37a) Knowledge of its fate subsequent to melt refining is required to establish specifications for the off-gas-handling facilities of the EBR-II Fuel Cycle Facility in order that release of iodine-131 to the atmosphere will not result in air concentrations in excess of maximum permissible air concentrations established by the AEC. Trice has found(40) that only about 10^{-3} percent of the iodine in the original irradiated material was removed from a melt refining furnace on pumping it down to a final pressure of 8 mm Hg with the melt refining crucible at a temperature of 650°C. Apparently, iodine is deposited on walls and surfaces of the melt refining furnace. In that investigation, a trap (which consisted of a bed of activated charcoal with AEC filters on the fore and aft sides of the charcoal) was inserted between the vacuum pump and the melt refining furnace to remove particulate material. The trap removed essentially all of the iodine activity from the gas stream and retained it at the low (8 mm) pressure existing at the end of furnace pumpdown. The effluent from the iodine trap contained in the order of 10^{-5} percent or less of the iodine in the charge.

Several of the volatilized elements are condensable (cesium, cadmium, and rubidium, as well as any sodium which may adhere to the pins after decanning). These materials will be removed in a trap placed over the melt refining crucibles.

Reactive Fission Products

The second or reactive group includes the fission product and transuranium elements listed below. It has been shown that at least 95 percent removal of these elements will occur in melt refining.(30, 36a, 37a)

Element	Symbol	Atomic Number	Weight Percent Formed Per Reactor Pass at 2-a/o Burnup
Strontium	Sr	38	0.076
Yttrium	Y	39	0.038
Tellurium	Te	52	0.026
Barium	Ba	56	0.080
Rare Earths	RE	57-63	0.625
		Total	0.845
Americium	Am	95	-
Thorium	Th	90	-

The total weight of these fission products formed per reactor pass is about 40 percent of the total fission product weight. Thorium, which might be introduced into the fuel from mold or crucible coatings,(41) and any americium generated in the fuel are expected to behave as reactive elements.

The rate of removal of cerium (as a standin for all rare earths) from EBR-II fuel alloys has been studied extensively.(30,35) It is expected that over 90 percent removal of these elements can be achieved at 1400°C within 3 hr in a lime-stabilized zirconia crucible. The predominant mechanism of their removal is by the formation of oxides at the wetted crucible surfaces, with oxygen contributed by the zirconia crucible. In general, the oxides remain in a reaction layer which is formed on the wetted crucible surfaces. The zirconia is left in an oxygen-deficient condition. Introduction of zirconium into the uranium melt is negligible.(35)

Noble Fission Products

The third or noble metals group for which practically no separation will occur in melt refining,(23,28,32,34,35,36a,37a) is comprised of the following elements:

Element	Symbol	Atomic Number	Weight Percent Formed Per Reactor Pass at 2-a/o Burnup
Zirconium	Zr	40	0.246
Niobium	Nb	41	0.013
Molybdenum	Mo	42	0.190
Technetium	Tc	43	0.055
Ruthenium	Ru	44	0.144
Rhodium	Rh	45	0.026
Palladium	Pd	46	0.015
Silver	Ag	47	0.001
Indium	In	49	0.0004
Tin	Sn	50	0.0036
Antimony	Sb	51	0.0025
		Total	0.697*
Neptunium	Np	93	-
Plutonium(28)	Pu	94	0.237
Uranium-236	U^{236}	92	0.450

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*Thirty-three percent of the total weight of fission products generated per reactor pass.

Studies have shown that zirconium can be removed in melt refining by scavenging with carbon,(32) but there are no present plans to remove zirconium by this procedure. Hampson has also observed that silicon (introduced, for example, from Vycor molds in the refabrication step) would also behave as a noble element.(42)

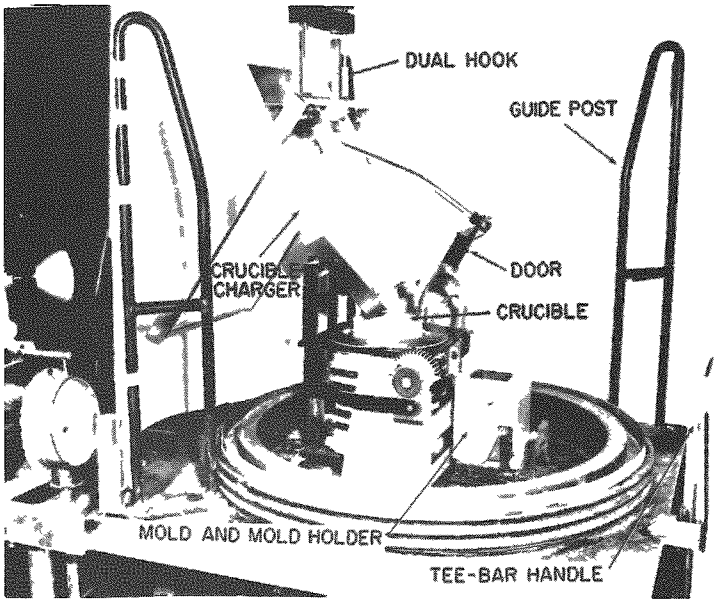
2.1.6.3 Equipment Design(43)

The melt refining furnace is induction heated and can handle a charge of about 10 kg. It is of the tilt-pour type and is mounted on a raised platform (see Figures 21 and 22).(4,37b,43,44) The furnace is covered by a removable bell-jar-type top which has a freeze-metal seal. A ceramic fume trap or filter covers the crucible during melting.(37b,45a)

The melting is done in a dry-pressed, five percent lime-stabilized zirconia crucible (Norton mix RZ300) which has a $5\frac{1}{8}$-in. inside diameter and $6\frac{3}{8}$-in. outside diameter, is $9\frac{1}{2}$ in. high, and $8\frac{1}{2}$ in. deep. The crucible fits loosely in a removable graphite inner susceptor which has a $6\frac{15}{32}$-in. inside diameter and is $7\frac{13}{16}$ in. deep. The inner susceptor is held, in turn, by a fixed, outer graphite susceptor which has a $6\frac{15}{16}$-in. inside diameter and is 8 in. deep. Both susceptors have $\frac{3}{16}$-in.-thick walls and bottoms. The outer susceptor rests on a 1-in.-thick zirconia insulating brick and $\frac{1}{4}$ in. of Fiberfrax insulation, which in turn rest on a $\frac{3}{8}$-in.-thick steel base. A 10-in. outside diameter by $8\frac{3}{4}$-in. long by $\frac{7}{8}$-in. thick molded Fiberfrax insulating cylinder rests on the base and surrounds the outer susceptor. The space between the susceptor and the Fiberfrax cylinder is filled

Figure 21

MELT REFINING FURNACE - BELL COVER OFF - CHARGING POSITION



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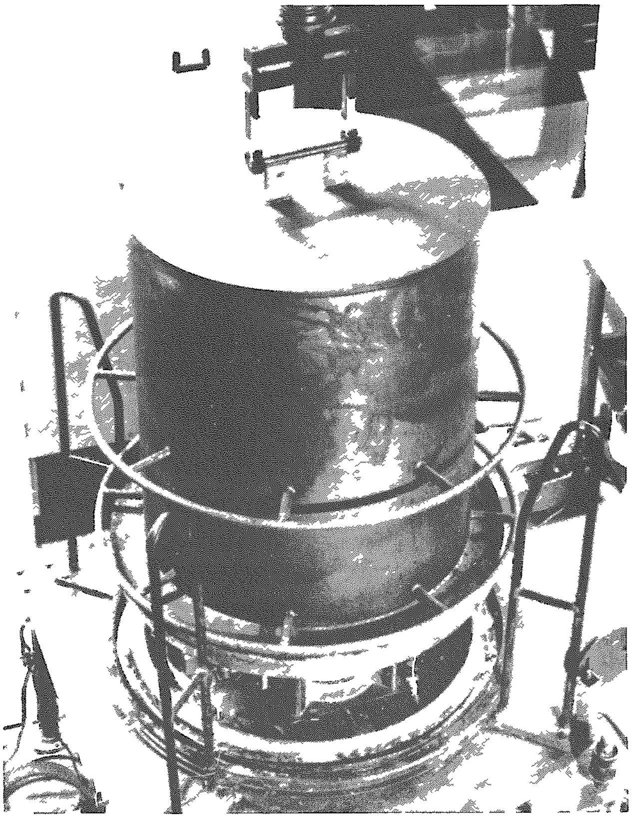
with silicon carbide insulating powder (approximately 800 mesh). The outer susceptor and insulation are held in place at the top by a zirconia insulating ring which is held by a steel ring. Figures 23 and 24 show the relationship of the various components within the melt refining furnace.

Figure 22

MELT REFINING FURNACE - BELL COVER BEING REMOVED



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A five-turn coil of $\frac{1}{2}$-in.-diameter copper rod surrounds the outer insulating cylinder for induction heating of the susceptor. The coil is about 5 in. high. To permit tilting, the coil is connected by woven flexible copper leads to power-input terminals on the furnace platform. An induction-heating frequency of 10,000 cps is used. A copper coaxial conductor with magnesia insulation extends from the furnace to a 30-kw generator and balancing capacitors below the argon cell.

The fume trap consists of a molded porous Fiberfrax* cup which has a 10-in. outside diameter and is $5\frac{1}{2}$ in. high with a $\frac{7}{8}$-in. wall (see Figures 23 and 24). In operation, the fume-trap cup is in an inverted position over the crucible, and its rim rests on and seals at the top steel ring of the crucible. The fume trap is held in a steel frame which automatically lifts it when the furnace is tilted to pour. The function of the fume trap is to condense and trap cesium and sodium vapors and any other condensable elements or compounds.

The receiver mold is made of graphite and is held in a steel container. The inside dimensions of the mold are: $3\frac{1}{8}$ in. square at the top, $2\frac{3}{4}$ in. square at the bottom, and $5\frac{1}{2}$ in. deep, with rounded corners. Ingot samples are provided by two tapered holes, about $\frac{1}{8}$ in. in diameter and $\frac{1}{4}$ in. deep, at the bottom of the molds. After the ingot is removed from the crucible, the two projections formed are broken off and used as samples.

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*A product of the Carborundum Co. The stated composition of the fibers is aluminum oxide, 51.2 percent; silicon dioxide, 47.4 percent; boron oxide, 0.7 percent; and sodium oxide, 0.7 percent.

Figure 23

MELT REFINING FURNACE - BELL COVER OFF - MOLD AND MOLD HOLDER AND FUME TRAP IN MELTING POSITION

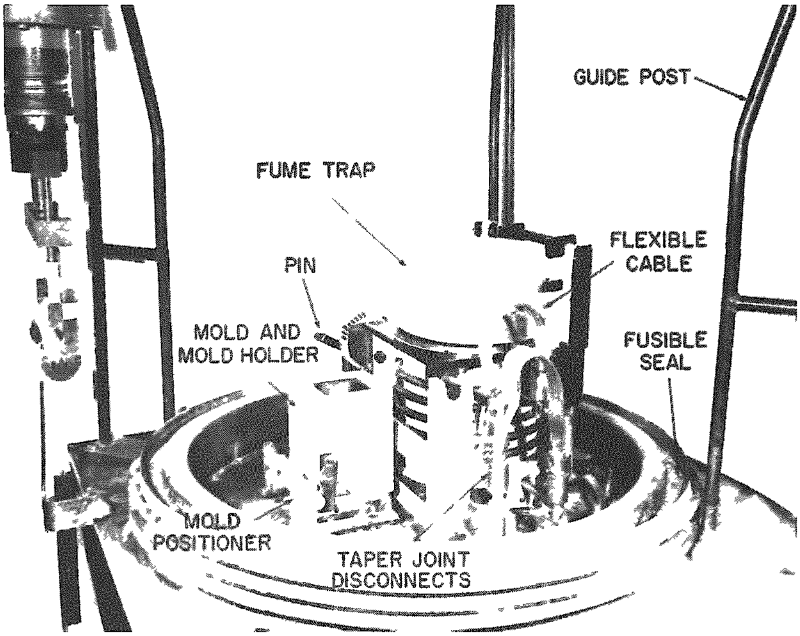

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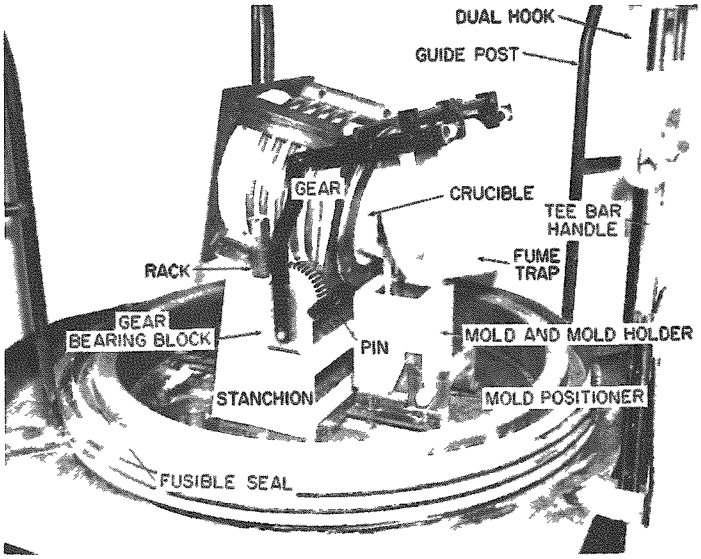
Figure 24

MELT REFINING FURNACE - BELL COVER OFF - POURING POSITION



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The bell jar cover (Figure 22) is made of steel. It is about 30 in. in diameter and 41 in. high, and its lower enlarged rim fits into a 37-in.-diameter freeze-metal seal trough located on the platform. The seal is heated by electric heaters for melting of the metal. The melting point of the metal used in mock-up tests is about 130°C, but metal with a higher melting point is to be used in the argon cell. The time required to set the bell jar and allow the seal to freeze is estimated to be $1\frac{1}{4}$ hr. The free volume inside the bell jar cover is about 16 ft^3.

The space within the bell jar will be evacuated and pressurized by vacuum pumps and argon equipment which are located outside the argon cell. A high-temperature, high-efficiency AEC absolute-type of filter is inserted between the furnace and vacuum pump. (It will possibly contain a charcoal trap for removal of iodine.) The flow of gases and fumes from melt refining is through the fume trap, through the high-efficiency filter, through the vacuum pump, to a negative-pressure holdup tank; at controlled intervals, the flow is from the holdup tank through a heater, a silver nitrate tower, a cooler, and then through a vacuum pump and a filter to the suspect stack.

The high-efficiency filter consists of an 8-in.-diameter by 10-in.-high cylinder with top side inlet and bottom outlet. It is incorporated in the vent line and held in position inside the argon cell on the furnace platform by means of metal freeze seals. The seals are thawed by means of clamp-on heaters carried by the manipulator. The filter connections are being designed so that the overall dimensions of the filter (with connections) are small enough to permit its disposal in a standard container.

A pin-charging apparatus (fuel-transfer pan) is used to store and move the chopped pins from the decanning apparatus and to deposit them in the melt refining crucible (see Figure 21). This apparatus consists of a closed rectangular steel pan 2 in. deep. One end of the pan tapers to an opening through which the chopped pins are loaded into the pan. The pan is 15 in. wide and 14 $\frac{1}{2}$ in. long up to the tapered section. The tapered section is $7\frac{1}{2}$ in. long and tapers to a width of $3\frac{1}{8}$ in., making the pan 22 in. long. The bottom of the pan is $\frac{1}{8}$-in. thick, and the sides and top are $\frac{1}{16}$-in. thick. The outlet of the pan is closed by a door on a hinged rod which automatically opens as the pan is lowered into pouring position above the crucible. The area of the bottom of the pan is about 280 in.^2. The projected surface area of 6070 g of pins (the weight of fuel in one subassembly) is 187 in.^2 when in a single layer. The intent of the pan design is to enable storage and handling of the pins in a single layer, insofar as possible, to minimize self-heating.

A reference chromel-alumel thermocouple, which extends through the bottom plate and bottom insulation to the bottom of the outer susceptor, is used to determine the temperature of the melt. The thermocouple, which is made of heavy sections of chromel and alumel for long life, is insulated electrically from the susceptor by boron nitride. It has wiping-type contacts to permit furnace tilting. The hot junction of the thermocouple is located about one inch above the bottom plate and $1\frac{9}{16}$ in. from the melt. This separation is made up of 1 in. of zirconia crucible, $\frac{3}{16}$ in. of inner susceptor, $\frac{3}{16}$ in. of outer susceptor, and $\frac{3}{16}$ in. of boron nitride. When the melt temperature is at 1400°C, this thermocouple indicates a temperature of 800 to 900°C, depending upon construction details. Individual calibration of each thermocouple installation with respect to actual melt temperatures is required. Work has been done to correlate reference thermocouple readings with melt temperatures for various amounts of internal electrical heating to simulate self-heating of the fuel by fission products.(36c)

When melt refining has been completed, the furnace is tilted by operation of an external lever with the manipulator. The fume trap is automatically lifted as the furnace tilts, and the melt is poured into the graphite mold (see Figure 24). The graphite-mold holder is in contact with a thermocouple which indicates by a temperature rise that the melt has poured. In past test work, tough slag films have occasionally formed over the melt and have prevented pouring. The possible need for a puncturing device has been considered, but none will be provided unless plant operation shows such a device to be needed.

After the ingot has cooled, the bell top is lifted, and the crucible, with skull, is removed and placed in a container for storage or skull oxidation. The ingot mold and holder are lifted and moved to a mechanical ingot-dumping station where the ingot is jarred from the mold. The bottom end of the ingot with the sample pins or projections is maneuvered into a sample-removing and catching device. A weight is dropped which breaks off the sample pins and knocks them through funnel-shaped holes into aluminum sample cans. The sample cans are closed by press-in lids and placed in a six-can holder for transfer to the air cell and from there to the analytical facilities. The ingot is then transferred to the pin-casting station or to temporary storage.

2.1.6.4 Self-heating

The self-heating of fuel with 2-a/o burnup in 135 days and 15-day cooling is shown in Table 5, page 21, for both 6070 g (one core subassembly) and for 10,000 g.

The self-heating effect of 6070 g of 2-a/o burnup chopped pins from one subassembly in the pin charger would be about 700 w for absorption of all beta and 25 percent of the gamma energies. The surface area of the pin charger is about 5 ft^2. If the pin charger is horizontal and the chopped pins in one layer on the bottom, the equilibrium temperatures for an argon cell temperature of 100°F (38°C) are estimated as about 640°F (350°C) for the pins and 365°F (185°C) for the charger surface. Temperatures may be even higher if pins pile one on top of the other. Provisions for cooling the pins in the charger while it is at the decanning machine may be necessary (see Figure 18, page 60). At this position, a fixed duct that carries cooling argon could be arranged to contact a connector on the charger. Possibly, the argon could flow through the filling opening to avoid the higher temperatures which might be produced due to the piling up of pins in the tilted charger. The effects of the sodium coating of the chopped pins upon the pouring of the pins from the charger into the melt refining crucible is not known. Due to self-heating effects, the sodium would be molten, and its presence may have considerable effect on the pouring behavior of pins.

As shown in Table 5, page 21, the total self-heating effect for a charge of 6070 g of 2-a/o burnup chopped pins from one subassembly in the melt refining crucible would be about 1220 w. Assuming that the pins [with a specific heat of 0.03 cal/(g)(°C)] and the 9600-g zirconia crucible [with a specific heat of 0.15 cal/(g)(°C)] are heated uniformly by this heat, the initial average heating rate would be about 0.17°C per second. In an hour or so, which is the estimated time required to replace the bell jar cover and freeze the seal, the average temperature might well be 1200°F (650°C), with the center of the pin mass being hottest. For a 10-kg charge, the temperatures would be higher.

The effects on the zirconia crucible of sodium present as a coating on the pins are somewhat unknown. However, it is known that molten sodium in direct contact with the crucible penetrates it and causes damage by weakening the structure. Due to self-heating, the pins may be warmer than the crucible before the furnace is turned on, and this could lead to the trickling of liquid sodium down onto and possibly into the crucible.

After the melt refining operation and on the basis of a 90 percent pour or yield (10 percent skull), the skull will contain fission products which are responsible for about two-thirds of the total self-heating(96) and the ingot will contain fission products which cause about one-third of the self-heating (ignoring self-heating of the volatilized material). (Actually, the fission products are distributed in three categories, approximately 43 percent in the skull, 27 percent volatilized, and 30 percent in the ingot.)

The isotopes zirconium-95 (half-life of 65 days), niobium-95 (half-life of 35 days), and ruthenium-103 (half-life of 40 days) contribute nearly all of the ingot-heating activity. On the basis that all of the beta and 80 percent of the gamma energies generated in the poured ingot are absorbed in it, the self-heating of a 5500-g ingot from one subassembly would be about 325 w.(17) For a 10,000-g ingot, the self-heating would be about 600 w. The 5500-g ingot would be about $2\frac{3}{4}$ in. square by $2\frac{1}{2}$ in. long; the 10,000-g ingot would be longer (total length of about 5 in.). In the argon cell, in either case, the equilibrium temperature of the ingot would be about 995°F (535°C) for natural cooling. At this temperature, excessive nitridation may occur if nitrogen is contained in the argon-cell atmosphere at the expected concentrations of several percent. A special container, possibly providing a pure argon purge or high-velocity cell atmosphere cooling, may be used for storing ingots. Such a storage arrangement could also be used for heels, shards, reject pins, etc., held as feed for the injection-casting operation.

Since the ingot in the pouring mold will be hot due to self-heating, it may be more difficult to dump the ingot from the mold than has been the case in work with unirradiated material. No experience is available on the dumping of relatively hot ingots (about 500° to 600°C) from molds. Furthermore, in experimental work, it has been noted that ingots tend to stick to degassed molds. The slow cooling of uranium in the mold due to fission product heating will allow degassing of the mold to occur and thereby further aggravate the problem of ingot sticking. Experience to date on use or re-use of molds has been with molds handled or stored in room air. Mold performance may be considerably different in the very dry argon atmosphere of the argon cell.

2.1.6.5 Summary of Operations

Each melt refining operation will require the following equipment items and feed materials:

1) Graphite pour mold (may be re-usable after visual inspection).

2) Zirconia crucible.

3) Fume trap.

4) Thermocouple (possibly will be required in addition to the reference thermocouple)

5) Decanned pins and (possibly) recycle scrap or make-up enriched uranium

The sequence of operations in melt refining is as follows:

1) Place a sound graphite mold in the mold holder and place the assembly in position on the furnace platform.

2) Place a weighed zirconia crucible in the suspector.

3) Put make-up uranium in the crucible, if required.

4) Charge the crucible with chopped pins (from the charging device) and scrap.

5) Place the fume trap in holder.

6) Set the ball jar cover in position, and freeze the seal.

7) Evacuate, and fill the bell jar cover with argon; repeat.

8) Melt the pins, liquate, and pour the melt.

9) After cooling, evacuate the bell jar atmosphere to the holdup tank, and vent the bell jar to the cell atmosphere.

10) Melt the freeze seal, and remove the bell jar cover.

11) Remove the fume trap, and place it in a container.

12) Remove the crucible with the skull, and place it in a container for storage or oxidation.

13) Move the mold to the demolding station, and remove the ingot.

14) Remove samples from the ingot, and place them in cans.

15) Transfer samples from the argon cell.

16) Transfer the ingot to storage or to the injection-casting station.

2.1.7 Skull Oxidation

2.1.7.1 Requirements

The purpose of skull oxidation is to oxidize the skulls from the melt refining operation so that the material can be removed from the zirconia crucible to which it normally adheres. After oxidation, the skull oxide can be poured from the crucible into a suitable container(46) for storage. It is easier (although still difficult) to get a sample from the skull oxide than from the skull itself. The skull oxide is starting material for the processing of skulls by liquid metal processes now being developed. (An alternative to complete skull oxidation is a partial oxidation which is sufficient to free the skull from the crucible. Following partial oxidation, the feed material to the skull reclamation process would be a mixture of skull oxide and unoxidized metal.)

2.1.7.2 Process

The skull oxidation process consists of first placing the melt refining crucible with skull in a stainless steel secondary can which holds the fragile and possibly cracked crucible together during the operation. The can with crucible and skull is placed in an oxidation furnace, where they are heated to about 700°C. The initial atmosphere in the furnace is argon. Oxygen is added at a controlled rate to limit the burning rate of the skull.(47) As the skull is oxidized, the reaction or burning rate decreases and the oxygen concentration may be increased. An automatic system is being developed for adding oxygen and venting argon so that the oxygen concentration is increased on the basis of pressure changes due to oxygen consumption.

The skulls can be oxidized at a rate of from 1 to 2 g of skull material per minute, which extrapolates to about 10 hr for a one-kilogram skull. After oxidation, the skull oxide is dumped into a container for storage or for charging into the apparatus for skull oxide reclamation. The particle size varies widely, but about 90 percent is between 325 and 14 mesh size.(46) Mechanical equipment for dumping the skull oxide is being developed.

Unpourable melt refining charges and slag materials also can be oxidized, if this is desirable, and recovered in the skull reclamation process.

2.1.7.3 Furnace

The skull oxidation furnace is shown in Figures 25 and 26.(48a) The furnace proper is about 15 in. high and 20 in. in diameter at the liquid metal freeze seal. A "Calrod"-type electric-heater coil

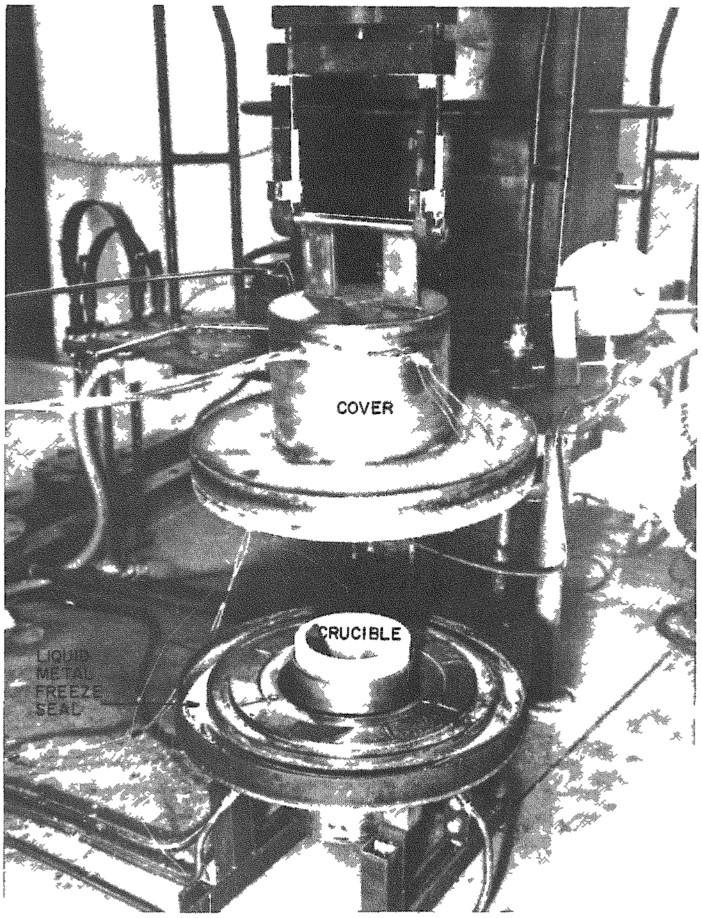

Figure 25

SKULL OXIDATION FURNACE - COVER OFF

108-4772

Figure 26

UNDERSIDE VIEW OF SKULL OXIDATION FURNACE

108-4781

(see Figure 26) of about 1.5-kw capacity is supported by the cover and enters the crucible when the cover is in place. This heater is employed to heat and maintain the skull material at about 700°C, which is the temperature at which the skull material is oxidized. A thermocouple in a well near the heater coil is used to control the electric current fed to the heater.

2.1.8 Injection Casting(41,49-51)

The first step in the refabrication of new fuel elements is injection casting of new fuel pins.(6,50,51) The purpose of injection casting is to convert the bulk fuel feed material into sound cylindrical castings, 0.144 in. in diameter and at least 15 in. in length, for producing 14.22-in.-long fuel pins for fuel elements.(41,6,50,51) The bulk fuel feed material consists of ingots from melt refining and scrap remelting operations (see page 112), and possibly of heels from preceding injection casting melts.

The basic method of this process is injection or pressure casting of molten fuel into precision-bore Vycor* (silica) tubes or molds to produce fuel-pin castings of finished diameter but of excess length.(52) In this casting operation, the fuel is melted in a crucible inside a closed furnace. Above the crucible, the Vycor molds are suspended vertically in a pallet. The tube molds are open at their lower ends and closed at their upper ends. When the melt is at the casting temperature (about 1300 to 1400°C for the first core loading), the furnace is evacuated, and then the crucible is raised until the molten fuel covers the lower ends of the molds. The furnace is then pressurized to about 25 psia, which forces the molten fuel up into the molds where it solidifies to form the castings. The crucible is lowered before the residual fuel or heel in the crucible freezes to the ends of the molds.

2.1.8.1 Equipment Design(41)

The injection-casting furnace (see Figures 27 and 28) is heated by a molybdenum induction coil (operated at 10 kc) which surrounds the crucible at its lower position. The thoria-coated graphite crucible rests on a zirconia insulator which is supported by an argon-pressure-operated pedestal which is raised about one foot for the casting operation. The crucible has a $6\frac{3}{4}$-in. outside diameter, a 6-in. inside diameter, is $5\frac{3}{4}$ in. high, and $3\frac{1}{2}$ in. deep. It is covered by a two-piece, hinged Inconel lid which automatically opens as the crucible is raised for the casting operation.

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*Vycor is the trade name for high-silica glass manufactured by Corning Glass Company.

Figure 27

INJECTION CASTING FURNACE

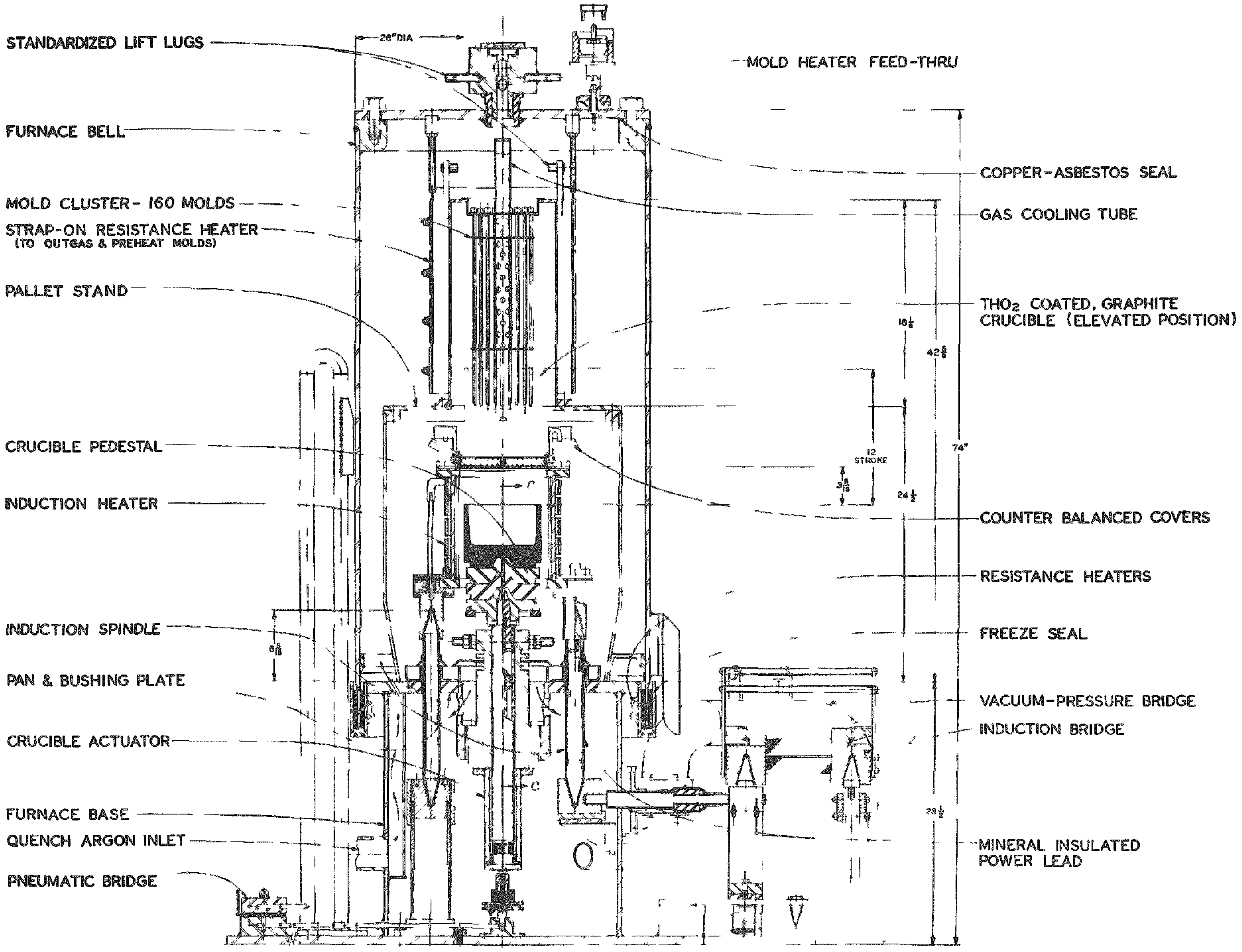
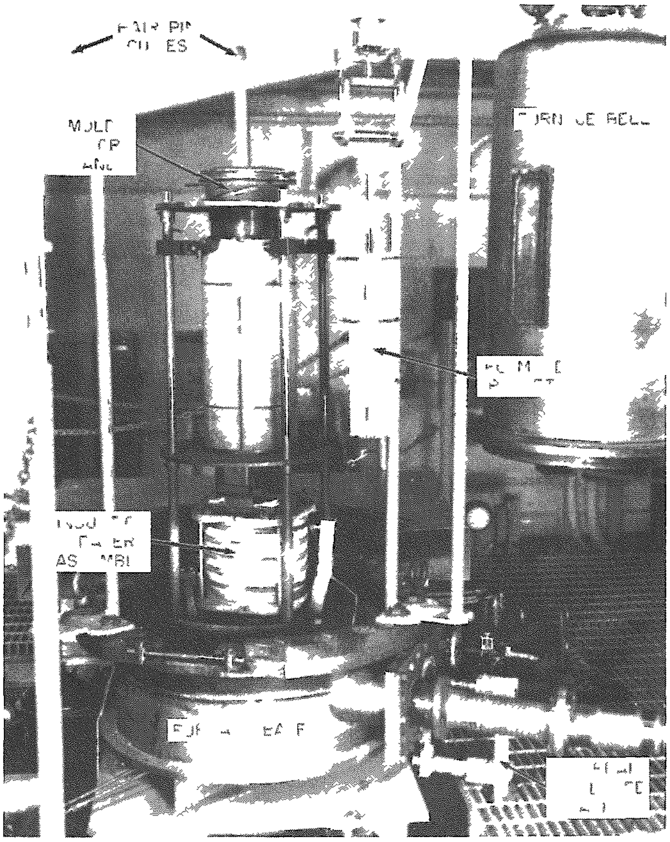


Figure 28

LOADED PALLET SUSPENDED BESIDE INJECTION CASTING FURNACE



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A normal charge consists of about $12\frac{1}{2}$ kg of fuel, which is sufficient for filling about 120 Vycor molds (charges up to 16 kg with 160 molds may also be utilized). The number of molds employed must match the charge weight to secure satisfactory freezing rates in the molds and crucible. The molds consist of Vycor tubes which have a $\frac{15}{64}$-in. outside diameter, a 0.148-in. inside diameter, and a $18\frac{3}{8}$-in. length. The upper ends are closed by flattening and fusing the upper $\frac{3}{8}$ in. of the tube; the lower ends have an outside bevel of 60° with the tube axis. The interiors of the tubes are coated with thoria. The lower several inches of the tube exteriors might be coated with aluminum oxide by flame spraying to prevent metal adherence. The molds are held in vertical alignment in a pallet which consists of a pair of 6-in.-diameter perforated plates welded to a center tube (see Figure 28). The Vycor molds are placed through the aligned perforations and are prevented from falling by the flattened section on their upper ends. The bottoms of the molds extend about $4\frac{1}{2}$ in. below the lower plate of the pallet. The pallet is placed on a support in the furnace so that, during heatup of the charge, the lower ends of the molds are above the crucible lid. A cooling system is arranged to circulate cold argon over the molds after the casting operation to hasten cooling and to insure freezing of the castings. This was not necessary for the casting of the nonirradiated fuel (no fission product heating) used for the first core loading. The furnace parts rest on a base and during injection casting they are covered by an elongated bell-jar-type cover (see Figure 28) having a liquid metal freeze seal at the bottom edge and which is guided into position by the hair-pin guides.

2.1.8.2 Detailed Operations

The first step in the injection-casting operation is the preparation of the charge and the transfer of consumable materials. Each injection-casting operation will require, in addition to the metal charge, the following equipment items:

1) thoria-coated graphite crucible;
2) pallet and Vycor molds;
3) thermocouple.

The preparation of the crucible and molds, described in Sections 2.6.4 and 2.6.6, includes heating of both of these to reduce the absorbed moisture. Since it is not feasible to seal these items hermetically during transport to the argon cell, these items must be expeditiously handled to reduce the probability of moisture absorption. Once in the argon cell, where the water content is expected to be between one and five parts per million, these items can be stored.

The Vycor molds can be loaded into a furnace pallet in either the air cell or the argon cell. Figure 28, page 78, shows a loaded pallet. The air cell has the advantage of greater (master-slave) manipulator availability. Loading in the argon cell reduces by one the air-lock transfers and also avoids a possible path for contamination of the air cell. Since the capacity of the standard transfer cart is a 10-in.-diameter, 21-in.-deep object and the loaded pallet is 6 in. in diameter and $20\frac{1}{8}$ in. long, a pallet can be accommodated by the small locks in a standard transfer cart.

The unprotected 4 to $4\frac{1}{2}$ in. of molds at the lower end of the pallet gives rise to a possible handling problem. If the pallet must be rested upright on a flat surface (as happens when it is placed in the transfer bucket of the air cell-to-argon cell transfer cart), initial contact is at the lower ends of the Vycor molds. The molds are free to move upward so that, when the pallet is lowered, the lower perforated plate supports the pallet. The unprotected molds are susceptible to damage when handled in this fashion.

The crucible is transferred to the loading station in the argon cell, where ingots and heels are added. A loading station is employed to preclude the dropping of fuel into the furnace and to offer the opportunity of inspecting the crucible for any damage during loading. A heel (that is, the material remaining in a crucible after injection casting) is removed from the crucible by turning the crucible upside down, broken, weighed, and stored in a steel container until used in the preparation of an injection-casting charge. The heel break will fracture the heel approximately into thirds. An average heel weighs 3200 g; one-third of a heel weighs 1100 g or approximately $2\frac{1}{2}$ lb. Because of this relatively large weight per segment of heel and because of the possibility of damaging a crucible by dropping such a segment, a manipulator will be used to transfer the large segment. The residue can be poured into the crucible. The charged crucible is placed on the crucible pedestal in the injection-casting furnace, and the pedestal is lowered into the heating zone. The mold pallet is then placed in position, and the furnace is closed.

The furnace charge for injection casting may be a controlled variable in the pin-processing cycle. It is not practical to expect an ingot of standard weight from melt refining. The figures shown on the fuel flow sheets discussed in the section on fuel movements and storage (Section 2.4, p. 137) are averages based on the experience with Core I (initial fuel loading) production, for which the feed material for the charge was fresh metal. There are two possible approaches to controlling the weight of charges to the injection-casting furnace. The first involves a backlog of melt refined ingots, remelt ingots (scrap which has been melted and cast into ingots), and heels. With an adequate inventory, the charge could be brought up to the weight necessary to cast a standard number of pins each time. An adequate inventory may, however, pose problems of criticality, self-heating, and nitridation of the feed materials by the cell atmosphere of argon - approximately five percent nitrogen. The alternative (and most probable) approach is to use the available feed material and make the number of molds the controlled variable. This approach appears to be more reasonable, but additional investigation of its effect on the casting cycle may be required.

The heel and ingot, once placed in the injection-casting crucible, constitute a heat source due to fission product decay heating. A calculation based on heat loss by radiation only shows that a crucible wall of a crucible containing a 10-kg ingot may reach a temperature of 715°F (380°C) and that a 10-kg ingot will have an equilibrium temperature of about 1200°F (650°C). Since it will not be possible to evacuate the furnace immediately after it is loaded with fuel, these temperatures may result in excessive nitridation of the charge.

For measuring the melt temperature, the furnace as designed includes a molybdenum-shielded platinum, platinum-10 percent rhodium thermocouple permanently placed and protruding from the crucible base plate. The hot junction of the thermocouple fits into a recess in the bottom of the crucible when the crucible is placed on the base plate. This thermocouple assembly rises with the crucible during casting and is, therefore, operable only when the crucible is in its lower position. Calibration curves relating the temperatures of the melt as read by an optical pyrometer and that measured by the recessed thermocouple have been developed.

In addition to the crucible thermocouple, connections in the furnace base plate for another thermocouple have been provided. These connections can be used in an attempt to employ a thermocouple in a tube in the mold pallet, which tube would be immersed in the melt when the crucible is raised.

Once the charge has been brought to casting temperature (~1300°C), the furnace is evacuated (to 60 to 100 μ) and the crucible is raised about 12 in., immersing the molds so that they nearly touch the bottom of the crucible. After a 4-sec delay, the furnace is pressurized. This forces or injects the molten metal up into the molds. Metal entering the bottom of the immersed molds solidifies upon immersion, and a 4-sec delay has been determined to be the time for the metal to liquefy again. Pressurization takes place at a rate of about 10 psi/sec to a final pressure of 20-27 psi. It has been shown experimentally that the rate of pressurization is a major parameter in controlling the casting process.(41)

After the metal is injected into the molds, it quickly solidifies from the top of the mold down to a point just above the metal remaining in the crucible. This chilling effect is temporary because of the heat reservoir of the crucible and still-molten heel. If left in contact with the heel, the cast metal begins to remelt upward. After a short time, solidification again occurs, proceeding downward from the top of the molten section until the entire unit - molds and heel - are solid.

It is critical in the casting operation to interrupt this cycle when the pins (castings) have been solidified to at least 15 in. of length and when the residual metal (heel) is still molten. From the above discussion, it can be seen that there are two points at which this separation can successfully occur. The first is when the pins are chilled after their initial pressurized surge into the molds (short cycle). The second is after resolidification of the metal in the molds has taken place and the residual metal (heel) is going through the transition from liquid to solid (long cycle).

The first or short-cycle method of separation was used for about 60 percent of the casting operations in the production of Core I. It has the disadvantage that any metal not solidified in the molds drips out; the method is also very sensitive to the initial casting temperature. It has the advantage of withdrawing the molds from a liquid heel, which reduces the possibility of bridging.* Since it relies on the chilling effect of the molds on the metal, the temperature of the molten heel does not indicate whether or not solidification has occurred. This method is now controlled by a time-delay cycle which has been determined for a furnace containing an average charge, and with the melt temperature read with an optical pyrometer. This method has the distinct disadvantage of being sensitive to self-heating. To counteract the effect of self-heating, the design of the remote furnace includes facilities to blow argon past the molds immediately after casting. This cooling tends to negate the self-heating effect (the proposed cooling rate is that provided by 17 cfm of argon at an initial temperature of about 80°F), but since calibration of this method with varying self-heating effects will not be possible in operations with nonirradiated fuel, the exact conditions for the short cycle will be very difficult to predict. At present, the cycle used is 12 sec (4 sec to heat the molds before injection and 8 sec for the castings to solidify initially).

The second (or long-cycle) method of casting is to await resolidification of the metal in the molds and, with the use of an immersion thermocouple, effect separation as the heel metal enters the mushy liquid-to-solid transition This method was used in producing about 40 percent of the castings for Core I, and in general has produced higher yields. Difficulties in utilizing this method in remote operations are in obtaining a thermocouple-protection tube that has the necessary fast response and that will not contaminate the fuel.

It has been shown that either of these methods is a practical approach to the terminal operation of the casting cycle, although neither has yet been used in remote operation. Part of the difficulty is that the short-cycle method must be calibrated to a specific furnace situation and that this calibration must include the unknown effect of the variable of self-heating. It should also be noted that concessions to design and procedures for remote operation have reduced the yield of good castings from ninety percent (in early experimental work) to the low eighty percent range. This constitutes an area in which a better understanding and control of the operation can be profitable.

A recent addition to the injection-casting furnace, which may have a pronounced effect on the casting cycle, is the use of "blast quenching." Blast quenching after casting is accomplished by blowing argon gas through the mold pallet from a perforated center tube The concept of blast quenching was initially introduced as a means of insuring

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*Bridging - a term used to describe solidified metal between two or more adjacent molds. In addition to increasing the metal loss, bridging creates a serious problem in the removal of molds from the pallet.

retention of enought of the soft, high-temperature gamma phase so that the shearing operation could be accomplished. Indicative of the effect of the phase transformation on the shearing operation is the relative hardness of the phases; alpha fissium (room-temperature stable) has a hardness range of 500-600 V.H.N. (Vickers Hardness Number), whereas gamma fissium has a hardness range of 125-175 V.H.N.

In addition to allowing the retention of some of the gamma phase, the use of blast quenching has been effective in reducing the variation in the fraction transformed from the gamma to alpha phase in a batch of cast pins. A variation in transformation leads to variations in the pin volume due to density differences in the phases (18.2 g/cm^3 for alpha and 18.0 g/cm^3 for gamma). These variations of pin volume, although not large enough to cause pin rejection on a dimensional basis, are large enough to affect the final sodium level. A change of 0.0001 in. in the diameter of the fuel pin causes a change of 0.026 in. in the sodium height.

The effect of blast quenching is to decrease the range of the cooling gradient in a batch of pins and to produce a group of pins whose degree (or percentage) of phase transformation is nearly the same from the center to the outer edge of the group of molds in the pallet. Additional discussion of the effects of blast quenching on the sodium level is included in Section 2.1.12, Bonding and Testing, page 102.

In the short-cycle separation, the quenching gas is introduced seconds after the metal is forced into the molds. In the long-cycle separation, the quenching gas is not started until after the crucible and molds have been separated. The flow of gas is from the bottom of the furnace to the top where it enters the center tube of the pallet. The center tube of the pallet is perforated, which allows the gas to flow radially through the cast pins and then down to the gas exit at the base of the furnace. Initial blast quenching has been successful in retaining enough of the gamma phase to insure ease of shearing and has a pronounced beneficial effect on control of sodium level.

When the crucible has been retracted and the cooling cycle completed, the metal seal is melted and the furnace bell-jar-type cover is removed. The pallet and filled molds are lifted by the manipulator and transferred to an unloading station. The crucible is removed and transferred to the heel break station.

A problem noted in the Core I production for which no countermeasure has been devised is the entrapment of molds that have dropped out of the pallet into the heel. Whether the top flange of the Vycor mold snaps off due to thermal shock or due to impact (high-speed movies have shown that upon pressurization molds are lifted by the upward flow of metal), the result is the same. The mold is free to drop into the heel, and at the time of separation of the mold and the heel, the broken mold or molds remain in the heel. A protruding portion of a mold and its contained metal can be broken off flush with the top of the heel by the master-slave manipulator. A section of Vycor tubing remaining in the heel has been removed manually (not applicable to the remote operation) or has been removed by drip casting or remelting.* The drip-casting method has also been used to refine heels that exhibit the buildup of a heavy dross which might prevent a good casting cycle.

The frequency of entrapped molds in heels is highly variable. It is normally zero, but has run as high as 20 molds per pallet of 120 molds. It has been estimated that five out of 100 casting operations would have entrapped molds. If one or more molds are entrapped in the heel, that heel will require further processing before it can be returned to the normal refabrication cycle.

One area of possible difficulty that has not been subject to test is the effect of gamma irradiation on the coated Vycor molds. The total exposure of the coated molds up to the time of casting has been estimated by Ayer and Feldman(53) to be 1.75×10^6 r (24 hr at 10^4 r/hr and 3 hr at 5×10^5 r/hr) at an average gamma energy of 0.7 Mev. The effect of the gamma irradiation on the thoria mold wash is not known. The effect on the bulk properties of the Vycor glass is probably minimized because of the annealing treatment to which the glass is subjected during the heating portion of the casting cycle.

In laboratory practice, the crucible is checked for loss of thoria coating after each casting cycle. If areas are chipped or cracked, the wash is repaired by spraying. In the cell, no facilities presently exist to respray the crucibles, and so disposal of the crucible after one use is planned. It may be possible to examine the crucible with the periscope and, if there is no loss of coating and no cracking, re-use may be possible. The possibility of respraying the crucible remotely has been considered, but the savings ($55 per crucible) would probably not justify the equipment and time that would be required.

Average yields were computed from the charges and products of 20 injection-casting runs. These 20 runs represent the

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*Drip casting or remelting - a procedure in which a double crucible is used. The upper section, containing the metal charge, has a perforated bottom. On melting, the charge is strained as it melts and passes through the perforations into the lower crucible. The lower crucible is of standard design. The major disadvantages of drip casting are the unrecoverable metal in the pores of the upper crucible and the fact that the lower crucible cannot be used directly for casting purposes because of the possibility of breakage on metal expansion.

final 25 percent of the Core I production. The average ingot weighed 9,290 g and a heel of 3,150 g was added to constitute an average charge of 12,440 g. The products of the casting runs averaged 9,037 g of cast metal, 3,247 g of heel, and 156 g of residue on the molds. Of the metal cast, 7,378 g were castings 15 in. or longer and 1,659 g were scrap (short or broken). Of the 15-in. or longer castings, 5,850 g were acceptable fuel pins, 656 g were not acceptable and were chopped, and 872 g of shards (pin ends) were produced in shearing. On an average, this indicates that 47 percent of the original charge (12,440 g) was produced as acceptable pins. In terms of "net pin yield" (acceptable pins/melt refining ingot), the yield is 63 percent. It is interesting to note that the average metal cast is 73 percent of the casting capacity. This is equivalent to filling 90 of the 120 molds. Finished pins were obtained from about 85 percent of the metal which was cast.

2.1.9 Pin Processing(54)

The purpose of the pin-processing operations is to produce finished fuel pins ready for the cladding operation. In the transition from raw castings to finished pins, the metal is removed from its mold, sheared to length, inspected for length, diameter, weight, and porosity, and stacked in an assembly-loading magazine. The byproducts of the process are waste molds and chopped fuel. The chopped fuel comes from both reject castings and the shearing operation.

The process must be capable of handling any possible form of product of the casting furnace and of rejecting all material not suitable for reactor use. It must also be designed so as not to introduce rejectable defects nor in any way reduce the output of the casting operation.

After the injection-casting operation, the pallet with filled molds is removed from the furnace and transferred to an unloading station, where the filled molds are removed from the pallet. The filled molds are then fed to the mold crusher where the Vycor molds are crushed and removed from the fuel-pin castings. Short castings and the mold pieces are separated from fuel-pin castings that are of the required length. The fuel-pin castings of sufficient length are moved to the pin shear, where they are sheared to the proper length of 14.22 in. The sheared fuel pins are then passed through inspection devices which weigh and measure the diameters and lengths of the pins. Those fuel pins which meet the specifications are placed in a loading magazine for assembly into fuel elements. Scrap fuel pieces, rejected pins, and mold pieces are collected for recycle or disposal.

2.1.9 1 Discussion of Process and Equipment

2.1.9.1.1 Pallet Unloading

The initial plans for the unloading stage contemplated the use of a fairly complex mechanical unit. These plans were abandoned and present operations are planned around the use of the Model A master-slave manipulator.

When the pallet is removed from the furnace it will be transported by the electromechanical manipulator back to the loading station (site of a pair of master-slave manipulators). The gravid or filled molds will be individually removed from the pallet and placed in the feed tray for moving to the pin-processing station. At this point, molds with obvious major casting defects, such as short or intermittent castings, can be removed.

2.1.9.1.2 Mold Crusher(54)

In the present design, the molds in transfer roll down a pair of spaced rails in the pin-processing machine (see Figure 29). Properly spacing the rail causes short pieces (less than 13 in. in length) to drop through the rails (be rejected) before they reach the crusher. Rejection of the short pieces also insures that no more than one piece at a time enters the crusher.

The crusher is a square-edged, recessed, horizontal slot that is large enough to pass a fuel pin but smaller than the mold (see Figure 30).(54) The gravid mold comes to rest over the slot. The fuel pin is forced through the slot by an overhead blade whose downward thrust breaks the Vycor and forces both the Vycor and the pin through the slot. Vycor and fuel are separated by interceptor rails just below the slot. The spaced rails carry the fuel pins to the next stage, whereas short fuel lengths and the Vycor fall into a collecting bin.

No provision has been made to wipe the fuel-pin surface after the Vycor is broken away. Two items of possible interest are the amount of fine Vycor carried with the pin and the amount of mold wash [98 percent ThO_2-2 percent kaolin (aluminum silicate)] that remains with the pin. Silica is not removed in the melt refining step.(42) The effect of the mold wash is probably not important as less than 0.25 g (half of the total used per mold) would be carried with each pin, and this amount of thoria is not expected to be deleterious to the process. The Vycor carryover may prove to be a more serious problem. An estimate of the silica carryover cannot be gleaned from the Core I production since,

Figure 29

MACHINE FOR FUEL-PIN MANUFACTURE(54)

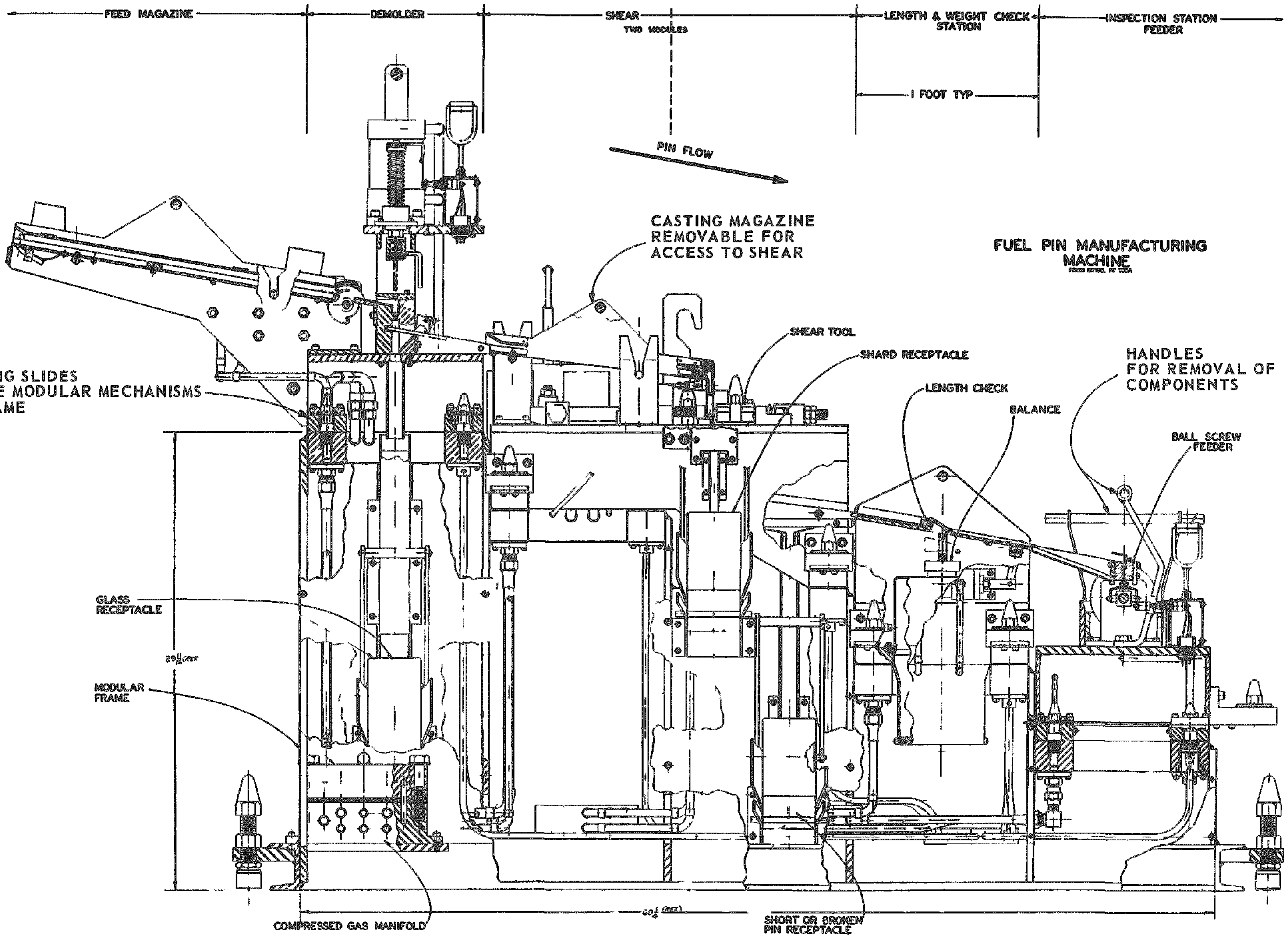

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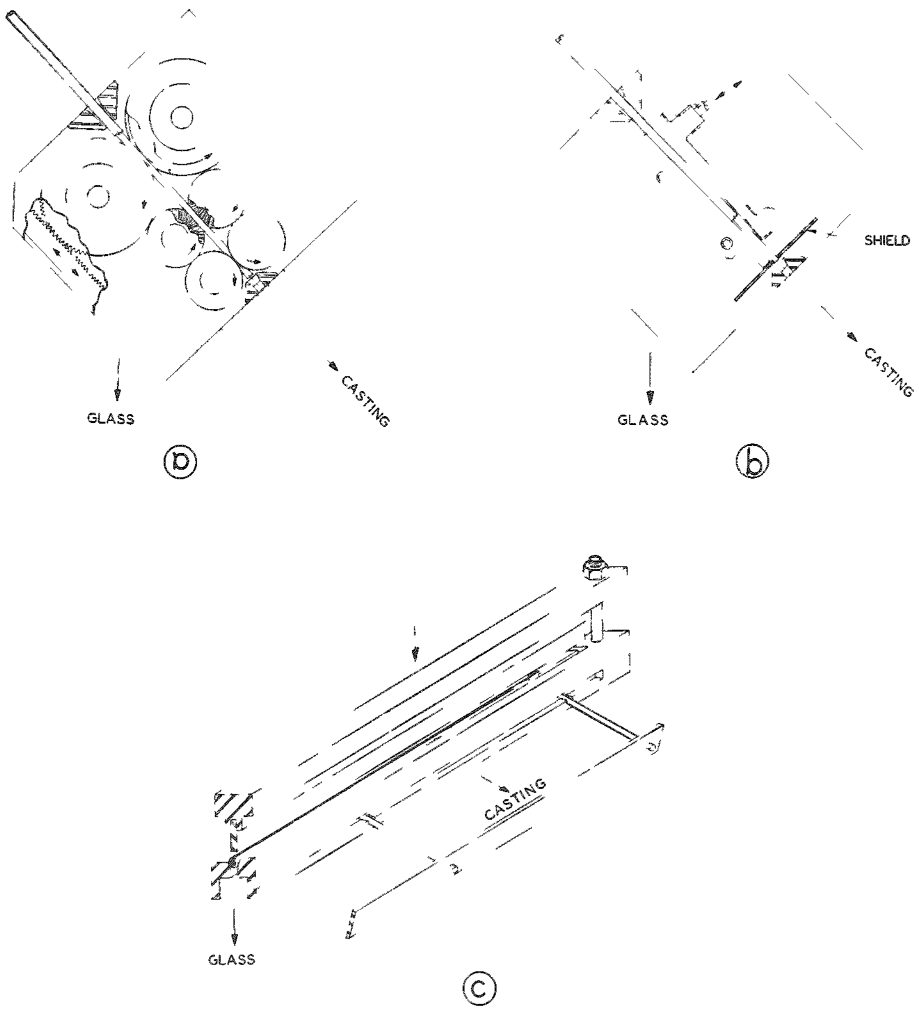
Figure 30

DEMOLDER FOR REMOVING FUEL PINS FROM VYCOR MOLDS(54)



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in that case, the pins were wiped clean (an action not reproduced in the proposed remote operation) and the crusher used was not of the type to be used in the remote operation. The effect of a silica build-up on the physical, casting and damage properties is discussed in the chapter on fuel metallurgy (Section 2.3.3.3.3, page 135).

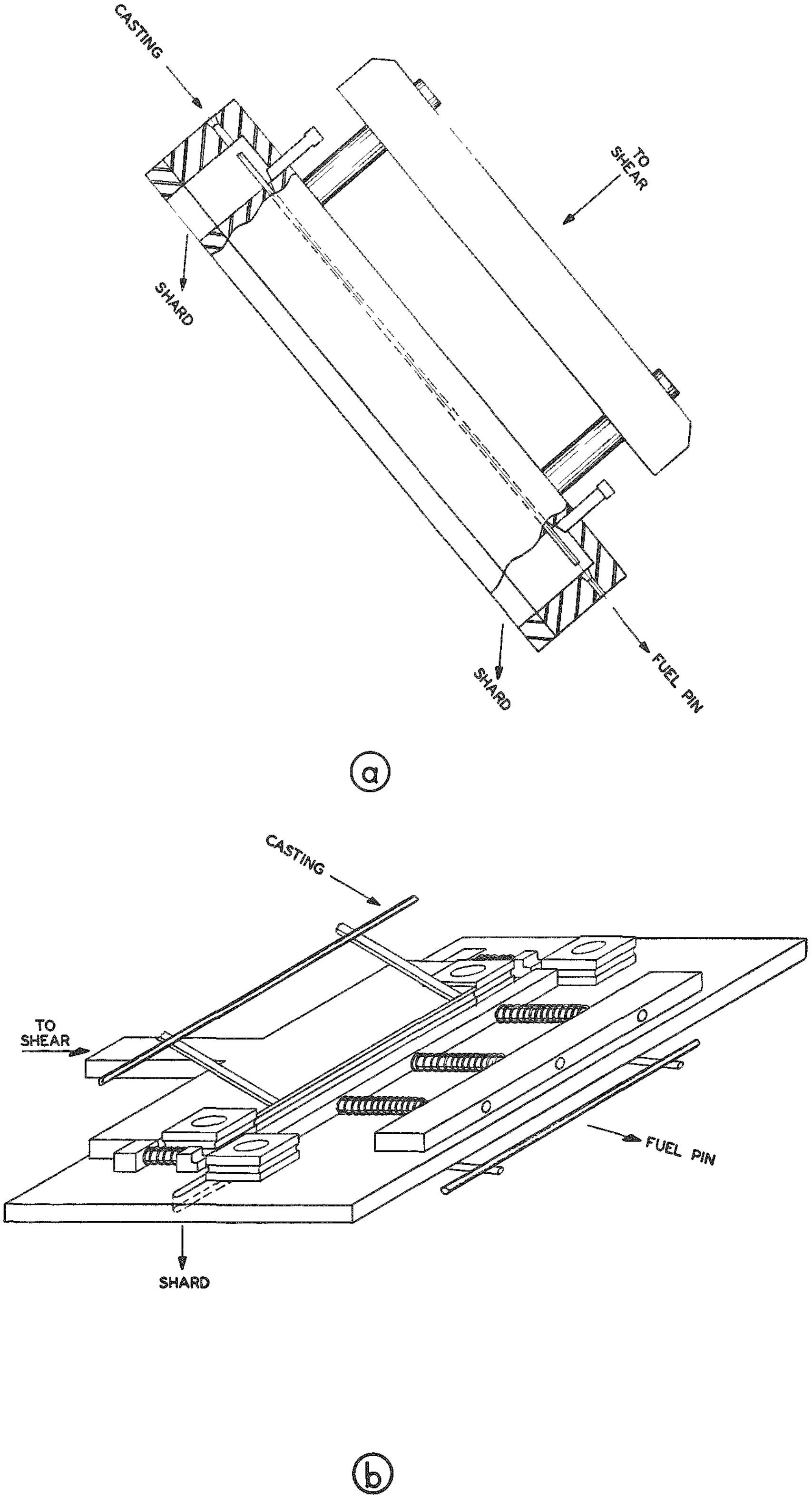
From the crusher, the pin feeds by gravity into a collection tray, and from the tray into the shear.

2.1.9.1.3 Pin Shear(54)

The pin shear cuts the pins to the required 14.22-in. length. The pin enters the shear from the top and is held between spring-loaded, horizontal, pressure pads and a pair of movable shear tools (see Figure 31). The shear tools (movable shear faces) are

Figure 31

FUEL PIN SHEAR(54)



operated by means of two pneumatic, cylinder-powered, bell-crank actuators. The pin is held by the three pressure pads. The shear faces are actuated, and the two ends are sheared simultaneously, leaving a center section 14.22 in. long. The movable shear faces are designed so that each unit has eight shear edges that can be remotely replaced or interchanged.

It is at the shearing operation that the effects of the phase transformations of the alloy are first noticed. The time-temperature dependence of phase transformations in the refined alloy may be slightly different from that for the laboratory variety of the same alloy, but the effect of the residual fission products should not make the following information irrelevant.

The fuel is blast quenched immediately after casting. This quenching (<u>sans</u> self-heating) has benefitted the process in two ways, one of which is pertinent here. The quenching has increased the percentage of cubic-structure (gamma) alloy retained in the fuel; it also appears to have reduced differences in the cooling rates throughout the filled molds and thereby reduced structural differences in the fuel pins of any batch. The percentage of retained gamma is important to the shearing operation since the phase transformation from gamma to alpha is accompanied by an increase from a hardness of approximately 150 V.H.N. (Vickers Hardness Number) to a hardness of 600 V.H.N. The alpha-transformed material is rather brittle (indicated by the 500-600 V.H.N.) and in this state is extremely difficult to shear to close dimensions. Experience gained during the production of Core I indicates that a 30 percent rejection rate due to the poor shearing qualities of the alpha-transformed material (out-of-tolerance fuel pins and fuel-end shattering) could be expected. In addition, the life of the shear blade would be sharply reduced if too little of the soft gamma phase were not present. The present shear is not designed to handle alpha-transformed material efficiently. For this and other reasons, it is imperative that, from injection casting through the bonding operation, the fuel be kept as cool as possible and not be allowed to approach the transformation temperatures.

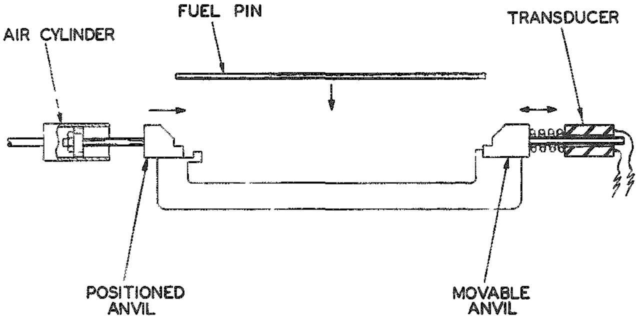
Since the quenched alloy is in a metastable state (the alpha phase is the room-temperature stable state), the partial retention of the gamma phase is a matter of controlling the length of time at temperature. It has been estimated that at 550°C the transformation to alpha takes 1 to 5 min, at 500°C it takes 5 to 10 min, at 375°C it takes 1 to 5 hr, and at 250°C it takes a time of the order of 100 hr (see data in Section 2.3.3.1, page 129). Obviously, the temperature of the cast fuel must be maintained below 250°C until all operations which depend upon the properties provided by partial retention of the gamma phase are complete.

2.1.9.1.4 Inspection Devices(54)

The sheared pins move by gravity from the shear to the length-and-weight-measuring station. The length is measured by pushing the fuel pin against a spring-loaded dimension-measuring transducer (see Figure 32). One end of the measuring device is accurately positioned; the other is spring-loaded and attached to the transducer. The deflection of the transducer is converted to pin length and recorded. The pin then drops onto the platform of a balance. The deflection of the balance is measured by a microformer, and the weight is recorded. The pin is elevated to remove it from the balance and is fed by gravity onto a longitudinal feeding device which moves it through the pneumatic pin-diameter gauge (see Figure 33) and an eddy current flaw detector (for surface defects or internal holes). Pin diameters may be measured accurately to 0.001 in. The information on the diameter measurement and flaw detection is recorded. The longitudinal feeding device then delivers the pin to an accept-reject station.

Figure 32

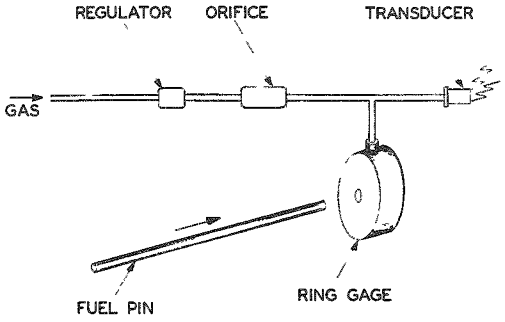
SCHEMATIC DRAWING OF LENGTH GAUGE(54)



106-5992

Figure 33

SCHEMATIC DRAWING OF PIN-DIAMETER GAUGE(54)



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With the pin at the accept-reject station, the test information - length, weight, diameter, and porosity - is accumulated and a judgment is made about the pin. To allow this decision to be made quickly on the intermittently moving stream of pins, the test information is fed to data-processing and recording equipment. The data-processing unit records the diameter (in units of 0.0001 in. versus axial location on the pin) and the porosity (dimensionless units versus location) in parallel traces. From the measurements of pin diameter, the area is integrated over the recorded length of the pin, yielding a calculated pin volume. In addition to the volume, the pin weight (expressed to the nearest 0.01 g) and the pin length (expressed to the nearest 0.001 in.) are also presented. The volume data is stored, and both the stored volume data and the calculated density of the pin are printed out. The storage capacity of the data-processing equipment allows all of the data to be presented in one accumulation, which makes a quick accept-reject decision possible by the operator. The acceptable pins are placed in a tray which is transferred to the loading magazine. The reject pins are chopped and returned to the cycle for remelting and reprocessing.

2.1.9.1.5 Scrap Handling

There are five scrap-producing areas in the pin-processing operation. The disposition of scrap from all of these areas has not been defined. The five areas are discussed below, and possible solutions to the dispostion of scrap are presented.

(1) Unloading Station

The manipulator-unloading step will serve as the first sorting operation in the inspection of the castings. At this stage, short, intermittent or broken castings will be rejected as scrap.

(2) Mold Crusher (demolder)

Fuel pins about 13 in. or longer are carried from the crusher by gravity-feed rails which are $12\frac{3}{8}$ in. apart. Fuel lengths shorter than about 13 in. and the broken glass are fed via a chute to a hopper.

The scrap hoppers from steps (1) and (2) contain in varying amounts:

(a) crushed Vycor molds;

(b) fuel encased in Vycor from 1 to 13 in. in length; and

(c) fuel rods from 1 to 13 in. in length.

The estimated amounts for a 120-pin casting batch are as follows:

(a) crushed Vycor - approximately 2250 g;

(b) encased fuel - 6-12 pieces from 1 to 13 in. in length; and

(c) bare fuel - 6-12 pieces from 1 to 13 in. in length.

In a normal production run, fuel material from items (b) and (c) above has weighed an average of 1 to 1.2 kg. This material, on poor casting runs, has weighed as much as 6.6 kg.

(3) The scrap generated at the shear is collected in two containers. One container receives a $\frac{3}{4}$-in. shear from the top end of the casting. The other container receives the cut-off bottom, which is up to 3 in. in length.

(4) Short pins or broken pins from the shear are collected in a fourth container.

(5) At the accept-reject station, the acceptable pins are placed in a tray for further transfer. The reject pins are placed in another tray.

Of the five scrap containers, only the two from the shear contain material that is in condition to re-enter the process directly. The other three present special problems. The first contains crushed Vycor molds, Vycor-encased fuel, and fuel pieces; the second contains fuel pin sections less than 13 in. long; the third contains rejects of standard length. The eventual path for all three of these fuel collections is consolidations by melting (see Section 2.1.14, page 112) and return to the process cycle.

The acceptable criteria for recycle of scrap are rather simple. This material must be clean and free of contaminants, and must be less than 3 in. in length (the mouth of the crucible charger is $2\frac{7}{8}$ in. wide).

Present plans for handling the scrap are based on the use of the master-slave manipulators, an auxiliary glass crusher, and an auxiliary fuel shear. Sections of glass-encased fuel can be placed in the auxiliary crusher to remove the glass. The product of this step can be combined with sorted lengths of scrap fuel and fed to the auxiliary chopper. The chopped material can be returned to the cycle. The separation of small pieces of metal from the glass scrap is a problem that has not been resolved. In need of resolution is the minimum size of pin fragment of fuel to be reclaimed from the scrap.

2.1.9.1.6 General

In the present plan of pin-processing operations (as described herein), there does not appear to be a self-heating problem. The fuel pins are individually handled in the processing steps (this is the minimum-temperature condition). Between processing steps, the pins are held in single-layer trays. A calculation of the temperature rise of 65 irradiated fuel pins in a single-layer tray gave an equilibrium temperature of 176°C (348°F).(17) The fuel in the scrap container can reach a higher temperature, but the temperature limitations imposed to retain the gamma phase are not important for such material. Nitridation of fuel by the cell atmosphere is not expected to be a problem at the low temperatures of pins during their processing.(55)

The pin-processing operations are the most mechanically complex steps to be attempted in the fuel-reprocessing cycle. They are also the operations for the which the greatest degree of automation has been attempted. Although intense efforts have been given to the major steps of the operation, the byproduct steps present a number of unsolved problems. In addition, little experience in the realm of remote operation has been accumulated for this portion of the process. In the production of Core I, which provided pseudo-remote experience for most of the steps in the proposed fuel cycle, these operations were carried out without providing valid tests of many of the proposed remote methods.

2.1.10 Canning of Fuel Pins(56)

The purpose of this operation is to can the acceptable fuel pins delivered from the pin-processing station to produce fuel elements. At the assembly station, the pins are placed in stainless steel cans, each of which contains a small amount of sodium. The filled cans are then taken to the welding station where a gas-tight closure of the stainless steel can is made. Feed materials for this operation are the accepted pins, stainless steel fuel-element cans (each containing about one gram of sodium), and an integral stainless steel top cap and spacer. The equipment involved is a mechanical loading device, a pin settler, and an electromechanical welding unit. The welded fuel elements are transferred from the welding station to the station for leak-testing operation in the air cell.

As delivered to the process, jacket subassemblies consist of a stainless steel tube, nominally 0.156 in. in ID with a 0.009-in. wall, to which is welded a spade fitting. The inside diameter of each jacket is measured to the nearest 0.00005 in. by averaging twenty measurements along the length of the jacket by means of an air gauge. Measured tubes are grouped into ten classes within the interval from 0.1555 in. to 0.1565 in.

A length of extruded sodium wire, weighed to ±2 mg, is inserted into each jacket. The amount is determined from the average diameter of the batch of pins being loaded and the class of jackets used.

The loaded jackets are heated to 150°C to allow the sodium to flow to the bottom. After cooling, they are transferred in a suitable container to the fuel rod assembly area in the argon cell.

2.1.10.1 Assembly(56)

The sodium-filled cladding cans and the acceptable fuel pins will be transferred to an assembly work station in the argon cell. The cladding cans will be placed in an "A" feed magazine (20-pin capacity), and the master-slave manipulator will be used to place an acceptable fuel pin in each of the cladding cans.

2.1.10.2 Settling of Pin in Stainless Steel Can(56)

The settling operation is a simple operation, intermediate between assembly and welding, which insures that the sodium is molten and that the fuel pin has positioned itself at the bottom of the can.

The magazine is placed in a shell furnace and heated to a temperature high enough to liquefy the sodium. The upper limit of temperature is the phase-transformation temperature of the alloy. The range of acceptable temperatures is large (from 97°C, the melting point of sodium, to 375°C, the upper limit of sluggish transformation to the alpha phase). In practice, a temperature of 250°C has been used. The problem of self-heating does pose a question of temperature control since the furnace, rather than the pins, is temperature controlled.

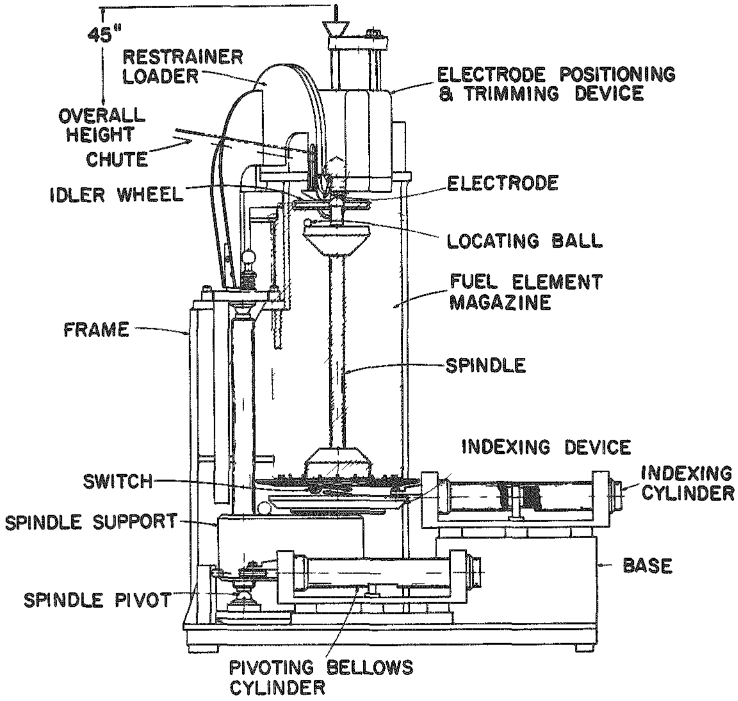
The temperature rise in a 50-pin magazine ("B" magazine) of similar design was calculated(17) to be 208°C. The rise in temperature in a 20-pin magazine will be less, as there is less fuel per volume of container in the 20-pin magazine than in the 50-pin magazine (ratio 1:1.2). The problem is to ascertain the temperature of the fuel in the presence of variable self-heating (the burnup will vary) when the temperature of the furnace rather than the temperature of the fuel element is being recorded.

2.1.10.3 Welding(56)

At the welding station (see Figure 34) two operations are accomplished: first, the addition of a top restrainer-cap to the assembled element, and, second, welding of the cap to the can. The station has for feed material an "A" magazine filled (or partially filled) with assembled elements and a hopper full of restrainer caps. Its product is a completed fuel element that is ready for final adjustment and test. The welding is accomplished by a shielded condenser discharge with a non-consumable electrode.

Figure 34

FUEL-ELEMENT WELDING STATION(56)



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The welding machine is designed to receive an "A" magazine filled with assembled pins. Since the welding gun operates in a vertical position and since the manipulator is not easily capable of a horizontal insertion motion, the welder is designed with a receiving station for the "A" magazine. The magazine, which has a ball-and-socket positioning fixture, is placed on the receiving device and is then swung into operating position. A cylinder-and-ratchet mechanism rotates the "A" magazine to expose the assembled elements to the welder.

The restrainers are fed from a hopper and are delivered to the assembly point in correct orientation for insertion. The grip mechanism, which places the restrainer into the fuel-element can, will not release a restrainer if it detects the absence of an element or the presence of an element already loaded with a restrainer. The spacer or restrainer is positioned to within $\frac{1}{16}$-in. of its final position before it is released from this station.

At this point, each of the 20 fuel elements in the magazine rests on a spring-loaded finger. As the magazine is rotated, each unit passes beneath a tapered ledge which fixes the height of the top of each assembly. As the units leave the wedge, they are held at this fixed height by cooling blocks pressed against the sides of the elements. The ledge and cooling blocks accomplish two important functions. First, for a unit which is acceptable for welding, they fix the welding gap at a constant distance. Second, if the restrainer is not correctly positioned, either too high or too low (or absent), the deflection of the spring-loaded finger opens the welding circuit so that it will not function. The rejected assembly must then be inspected to determine the cause of rejection. This mechanical arrangement reduces the number of faulty welds caused by incorrect positioning.

The welding cycle includes welding with a condenser-discharge arc in a helium atmosphere. The welder electrode is tungsten and is of the nonconsumable type. Since it does undergo alteration or wear in use, provisions have been incorporated to shear and reposition the welding tip remotely. Experience in the production of Core I fuel elements indicates that an electrode will weld between 100 and 200 elements before servicing is required. An automatic trimming operation removes $\frac{3}{8}$ in. from the electrode. The initial electrode length is 7 in., and the maximum usable length is $3\frac{1}{2}$ in. If, for purposes of preventive maintenance, the number of elements welded per tip is held to 100, the useful life of an electrode will be between 500 and 600 welds. This is of the order of two weeks of fuel-element production at the rate of 273 elements per week. The electrode is replaced by trimming the remaining $3\frac{1}{2}$ in. to scrap and inserting a new 7-in. section of tungsten. Insertion through an open, tapered hole is to be done by the electromechanical manipulator.

Welding is done in a helium stream. To date, experience with argon has not been successful; corrections to the electrical circuit to compensate for the difference in ionization potential and arc characteristics of the argon have not been determined. An estimate of the amount of helium used per 20-pin batch is 900 in.^{3} or about 3700 in.^{3}/day at standard conditions (the total amount of helium in Core I production was not measured). Since an estimated 300 ft^3 of fresh argon is to be added to the argon cell each day, the helium concentration in the cell will stay below $\frac{1}{2}$ percent.

Unless the welding cycle is interrupted by an indication of a reject operation, the cycle will continue until the entire 20-pin magazine is welded. The Core I production was an attended operation, and little experience in automatic operation was gained. It should be added that there are very few visible effects which are indicative of welding malfunction. Since it is not practical to remove the "A" magazine after each welding operation and observe the product, it will be possible to produce a whole magazine of bad welds. A better understanding of the electrical impulse used for welding could conceivably allow the use of a metered circuit that would indicate when a faulty weld was being made.

From the welding station, the magazine is transferred to the air lock and through to the leak-detection station in the air cell.

2.1.11 Detection of Leaks in Canned Fuel Pins(57)

The purpose of the leak-detection operations is to insure that the canned fuel pin or fuel element is gastight. This operation has the product of the welding machine as feed material, and this product is sent to the bonding station. The integrity of the fuel-element cladding insures that, aside from rupture or failure of the can, the fission products of the fuel element will be contained within the cladding and will not contaminate the primary sodium coolant in the reactor. The necessity for detection of fine leaks of the fuel element is based on the fact that a partially sealed fuel element can be seriously damaged by net sodium flow into or out of the element.

The leak-detection station is the first of the final stages of fuel-element production. As the element leaves the argon cell, it has undergone all of the required manufacturing steps. In the air cell, it is subjected to a series of tests which determine its acceptability for reactor insertion. Minor adjustments, such as rebonding, may make some rejected pins acceptable.

The EBR-II fuel element is designed with a gas void above the fuel and sodium. This void compensates for sodium expansion upon heating(58) and also provides space to accumulate fission gases that may be released. Sodium expansion in the fuel element is caused by the environmental temperature changes from 100°F at the time of manufacture to 700°F in the subassembly storage rack and to between 900°F and 1150°F when the reactor is at power.

If there were a small leak in a fuel-element can, sodium could flow into or out of the fuel element by a mechanism known as "sodium pumping." Some of the sodium or gas would be forced out of the element when it was heated up. Since the element is sealed at atmospheric pressure, the increase in pressure upon heating would result in sodium or gas flow to establish a simple pressure equalization between the fuel element and its surroundings. During reactor operation, this equilibrium condition would have no deleterious effect. Upon reactor shutdown, however, the pressure equilibrium would be disturbed as the fuel element cooled and came into a temperature equilibrium with the surroundings; there would consequently be a decrease in internal pressure. The pressure equilibrium would be re-established by a flow of sodium from the surroundings into the fuel element. Depending upon the size of the hole, the rate of temperature rise of a fuel element at reactor startup, and the amount of compressible gas left in the element at the high temperature-pressure equilibrium, a point could be reached where the exit of sodium from the element at high temperature rise would be so restricted by the hole size that the elastic limits of the cladding envelope would be exceeded and the fuel element deformed. Extreme pressure rise results in a fracturing of the cladding. If the hole was large enough to allow pressure equilization to occur, the damage caused by restricted flow would not take place, but escape of fission products would be probable. Since this reactor is designed on the basis that the fission products will be restricted to the fuel element, the integrity of the envelope (i.e., the can) must be ensured. If sodium were lost from the element and replaced by gas, melting or alloying of the cladding could occur due to absence of the sodium heat transfer bond.

2.1.11.1 Design(57)

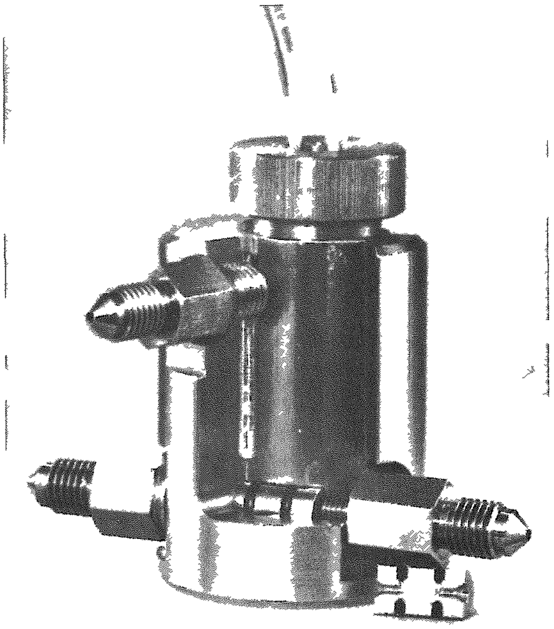
Initial plans for leak testing included a standard system for helium detection and the development of a leak-detection method that would successfully detect large as well as small leaks. Initially, leak detecting was accomplished by pressurizing with helium a closed vessel which contained welded fuel elements. Then the container was flushed with air and evacuated, and any helium leaking out of a fuel element was detected by a helium mass spectrometer. Since the container had been pressurized with helium, it was necessary to flush the system with air before the evacuation step could successfully detect helium leaking out of a fuel element. If the leak was large (comparatively), the leakage rate at normal atmospheric pressure was large enough so that under a vacuum all that was recorded was an initial spurt of helium.

To overcome this initial inability to detect large leaks with certainty, another method of leak detection has been developed. This method involves sealing the top one-half inch of a fuel element in a container of known volume (see Figure 35). This top one-half inch contains the weld, the fuel cans having been leak tested before they were placed in the cell. A metered amount of gas (helium) is forced into the known volume, and the resulting pressure is recorded. If the weld is tight, the metered volume of gas in the calibrated cavity produces a known pressure which remains constant. If the element has a defective weld, the pressure drops below the original value to one corresponding to the sum of the volumes of the cavity and the gas void in the element. If the pressure continues to drop to near zero, then a leaking test chamber is indicated and the test is repeated. Experience with this type of leak test has permitted the operators to correlate roughly the size of the leak with the rate of pressure drop. This method of leak detection has been successful enough to be adopted for all levels of leak detection (replacing the helium mass-spectrometric method and thereby reducing the number of leak detection operations).

Figure 35

FUEL-PIN LEAK TESTER(57)

(a) Leak Tester Head Valve Portion



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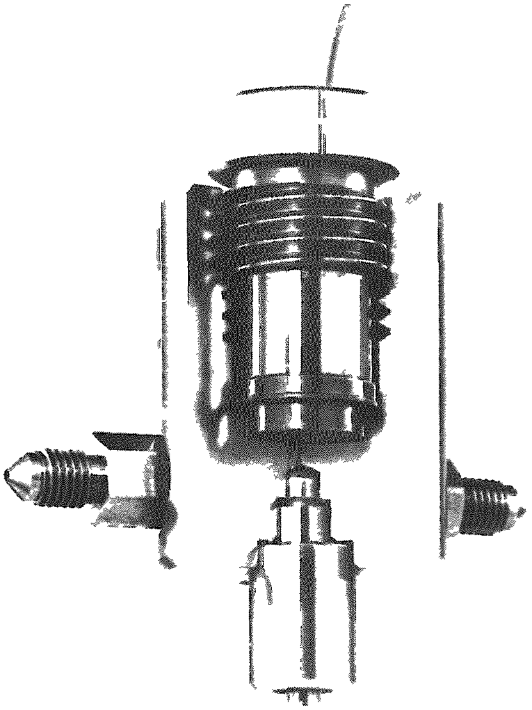
(b) Leak Tester Head Testing Portion



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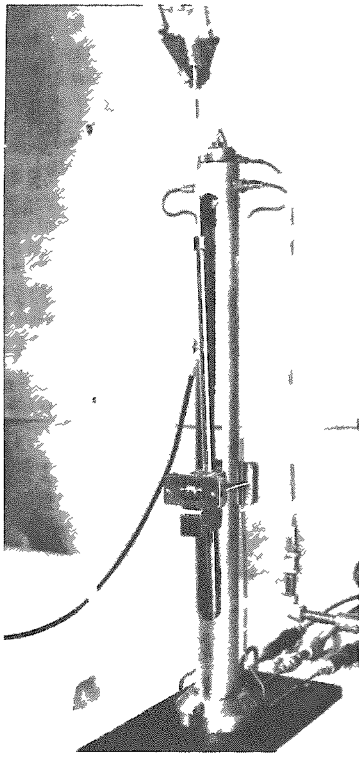

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(c) Overall View of
Leak Tester

2.1.11.2 Operation

The leak-detection station as designed for the Idaho operation includes five testing units and three stacking areas. The first stacking area is for an "A" magazine (feed), the second is for a "B" magazine (acceptable product), and the third is a feed tray for the decanning machine (reject product). The entire testing station is built into a table so that the work can be carried on at normal working height.

As there is a delay time to allow the pressure to come to equilibrium, it was valid to design the station as a multiple-unit stand. The pressure leak-test station will consist of five testing chambers. In the loading position, the lower chamber, where the fuel element is placed, faces the operator (see Figure 35c). The angle of forward tilt of the loading chamber is equal to the angle of the manipulator arm, which allows the fuel element to be lowered into the chamber with a simple manipulator motion. A circular gasket placed over the upper end (weld end) of the element is pushed down so that it rests on the upper end of the lower chamber. The test unit is activated, and the loading chamber is moved to a vertical position by cam action and then upward to mate with the test chamber. This is the test position. The upward motion of the loading chamber against the test chamber first positions the fuel element against the top of the test chamber (the lower end of the element rests on a spring base to allow for variation of element height) and, secondly, compresses the gasket, sealing both around the fuel element and at the lower flange of the test chamber. The test chamber is then pressurized (with helium) through the metering valve. The pressure at the metering valve is 1000 psi, and the volume of the valve is 0.004 in.^{3}. The volume of the test chamber (with a fuel element in place) is 0.0055 in.^{3}. so that when the metering valve is opened, the system pressure becomes about 420 psi. The void space in an element is approximately 0.03 in.^{3}. Thus, if a leak exists, the resulting pressure is approximately 100 psi. A pressure transducer is used as the sensing element in the system.

For all but the most minute leaks, a period of 10 min is adequate to make a determination. For very slow changes of pressure, longer times are necessary to determine whether the fuel element or the system is at fault.

There are two methods of handling fuel elements rejected at the leak-detection station. These are (1) rewelding or (2) recanning. Pins may be decanned and recanned if not damaged. Damaged pins would be recycled to melt refining. In the case of minor leaks, some success has been achieved (in Core I production) by subjecting the element to rewelding, although the effectiveness of rewelding is open to question. Very stringent requirements have been established for the welding operation, particularly with reference to the physical make-up and position of the parts to be welded. Since an element to be rewelded may not meet these requirements, there is a question as to the ability of the rewelding operation to correct deficiencies in the weld. There are three major types of weld defects:

a) contamination of weld area with sodium;
b) faulty centering;
c) cold weld (spacing or power input failures).

The first two defects do not respond to rewelding; the third may, if the welding machine has been readjusted. The rejection rate for leaking pins during Core I product was about 4 percent. It would appear that a rewelding operation should be tried, since the investment in time to reweld is much less than for the alternative of decanning the element and retrieving the fuel pin for reassembly. There is no evidence that subjecting a fuel element to rewelding more than one time has any value.

Since leak detection is accomplished in the air cell, the problem of manipulation is simplified. The welded elements are transported to the leak-detection station in the air cell in a 20-element "A" magazine, because the pressure-type leak detection is done on individual elements; unloading of the "A" magazine is an integral part of the operation. The welded elements from the pressure test are placed in a "B" magazine (similar to the "A" magazine except that its capacity is 50 fuel elements) for transport to the bonding station. The "B" magazine is a cylindrical magazine housing 50 elements in a single peripheral row. Transport of the "B" magazine from the pressure-leak-detection station to the bonding furnace is accomplished by a bridge manipulator through the use of a ball-and-collet compliant lifting tool. Master-slave manipulators in the air cell are subsequently used in the operation of the bonding and bond-testing equipment.

Calculations indicate that in the helium leak-detection operation, in which 10 pins are grouped together, the temperature rise under the most adverse conditions will be 360°C.(17) In the pressure chamber, only the upper portion of the element (nonfuel-bearing portion) is confined in a vacuum, and so a temperature problem does not exist. The temperature rise in the lower portion of the fuel element surrounded by an enclosure but subject to convective cooling was calculated to be 37°F. The temperature rise for elements in a full "B" magazine has been calculated to be about 330°F.(17) If necessary, forced cooling can be used to reduce the temperature.

2.1.12 Sodium Bonding and Bond Testing(56)

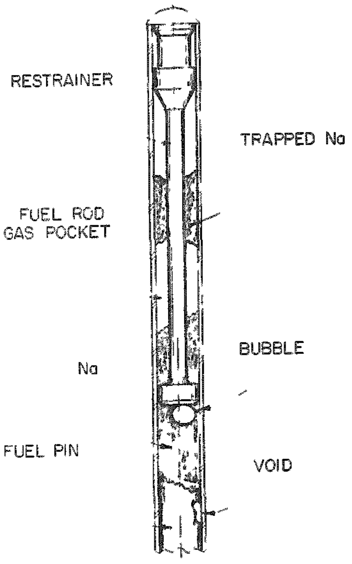
The purpose of the bonding and bond-testing operations is to insure that sodium-fuel and sodium-cladding bonds have been optimized. The sodium envelope which surrounds the fuel pin is primarily a heat transfer medium. The sodium level must extend high enough to insure coverage of the fuel as it expands longitudinally and yet not so high as to impair operation of the compression cushion of the gas void. The finished height of the sodium is checked to insure that it lies within acceptable levels. The feed material for the bonding and bond-testing operations is leak-tested fuel elements, and the product is fuel elements ready to be assembled into reactor subassemblies.

2.1.12.1 Bonding(56,59)

The insertion of the fuel pin into the sodium-filled can or jacket and the subsequent settling operation are primarily positioning steps. Because the fuel element is not in its finished condition at that time, it was thought that the actual bonding and testing operations should await a time when no further work would disturb the established sodium bond or level.

Figure 36

SODIUM DEFECTS IN FUEL ROD(56)



112-703

Three fuel pin defects can be corrected in the bonding operation (see Figure 36). The most important defects are gas bubbles or voids trapped in the sodium, particularly on either of the bond interfaces. It is necessary that any entrapped gas in the sodium bond be moved to the void space at the top of the fuel element. The second defect is sodium that has been left on the inner jacket wall above the normal sodium level, and the third defect is a gas pocket which sometimes forms just under the end of the top spacer.

The methods which have been attempted to remedy these defects include high-frequency vibration, low-frequency vibration, and centrifuging. All of these methods were employed at elevated temperatures. The initial core loading was made using 60-cycle vibration and temperatures of about 500-550°C. Nonacceptable pins were reclaimed by a high-temperature centrifuging operation. The centrifuging operation was not deemed feasible for in-cell operation as the large space required and the problem of controlled heating of an in-cell centrifuge appeared to necessitate too large an investment for a secondary operation.

Reject rates related to bonding during Core I production were as follows:

a. Voids or bubbles = 7 percent;
b. Trapped sodium = 2 percent;
c. Spacer bubbles = 1 percent.

Near the end of the Core I production run, two studies of the bonding problem indicated a need for changes in the tests for bonding. The first study was an investigation of the possibility that some of the bubbles noted in inspection were not actually gas bubbles but were shrinkage voids. The second study was concerned with the effects of vibration frequency and amplitude on the separation of gas bubbles.

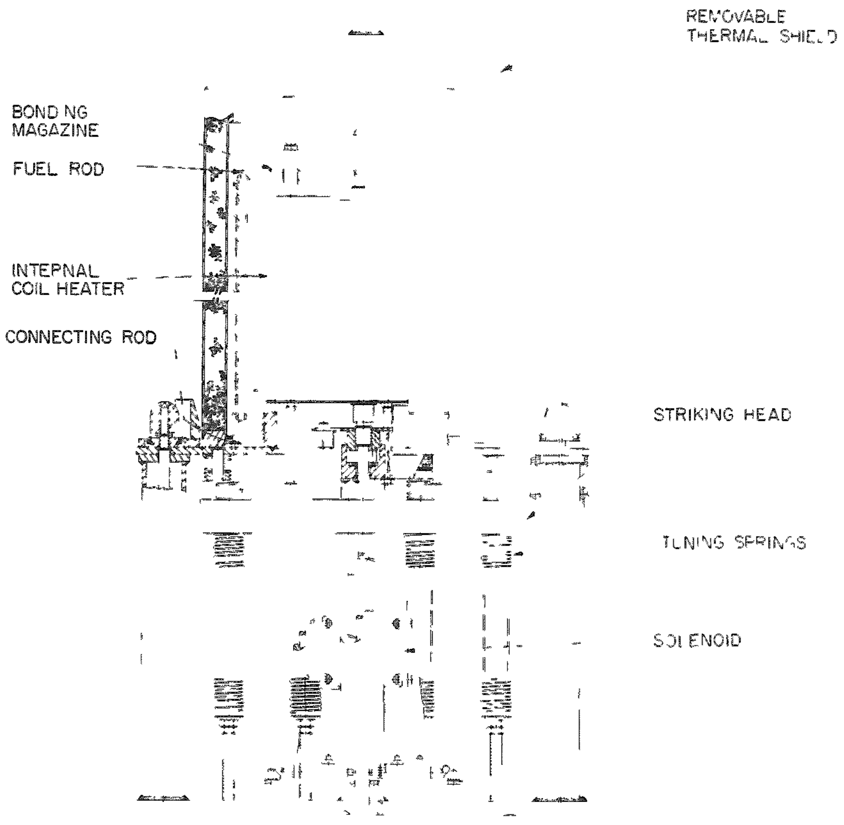
The possibility that gas bubbles should be differentiated from shrinkage voids (produced upon solidification) arose because production testing was done with fuel elements which contained solidified sodium, whereas at operating conditions (700°F minimum) the sodium would be liquid. Experiments were run in which elements rejected in the normal tests (with sodium solidified) were retested with the sodium in the liquid state. In addition, a series of tests were run in which voids or bubbles in a fuel element were recorded as a function of temperature. This last series proved rather conclusively that internal shrinkage voids had existed and that they had been a cause of rejection in the normal test at room temperature. These studies have led to the conclusion that acceptance tests for sodium voids and bubbles should be run at temperatures above the melting point of sodium.

In the second study, which was based on observations made during production bonding, the effect on bonding of low-frequency, high-amplitude motion was investigated. The experimental machines, called jolters or thumpers, impart an upward motion to the elements and then retract, allowing the elements to fall back into their receptacles. The force of the return impact is used to densify the sodium-bonding layer. The good results obtained when low-frequency, high-amplitude bonding motion was used warranted a design change which incorporates these operating characteristics into the machines for the Idaho operation (see Figure 37). This approach to bonding is also effective in shaking down sodium trapped in the void space and freeing bubbles trapped under the spacer.

No individual fuel-element handling is required at the bonding furnace. The magazine is placed in position, and the heating and vibrating cycles are initiated. In the bonding operation, the spade end (lower end, see Figure 4) of the fuel element is housed in a slotted cup. This cup is designed so that it transmits the mechanical force of the bonding vibration to the end of the cladding tube rather than to the spade itself. This is done to protect the spade from possible damage during processing.

Figure 37

SODIUM-BONDING MACHINE(56)



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2.1.12.2 Bond Testing(56)

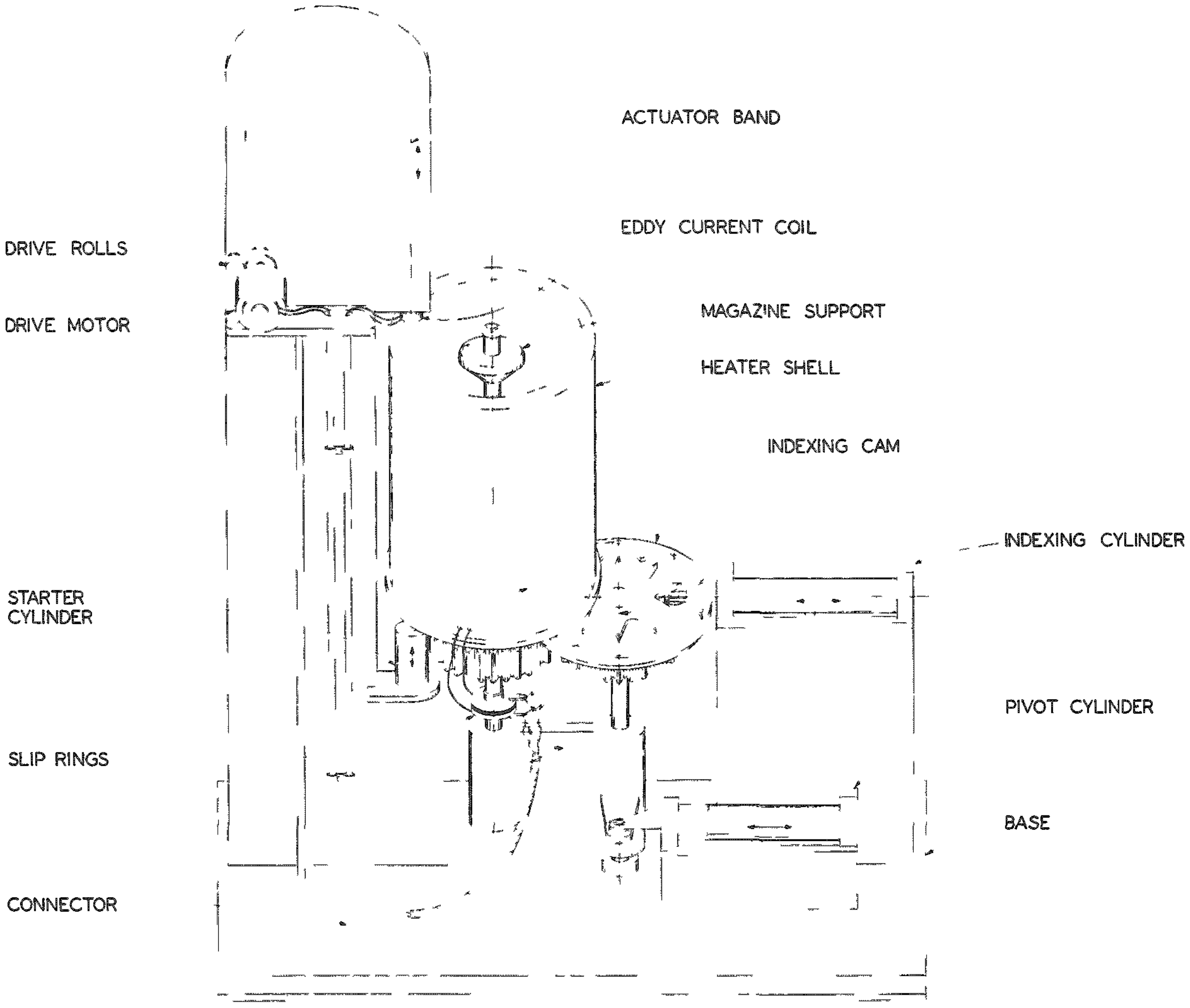
The detection of voids in the sodium bond area is accomplished with an encircling-coil eddy current probe (see Figure 38). Voids in the sodium induce changes in the coil impedance and therefore in the coil current. The remotely operated machine consists of (a) a pivoted loading and indexing platform capable of resuming the bonding magazine and keeping the sodium in a liquid state during inspection, (b) an indexing device for indexing the magazine and placing the fuel rod in position under the coil, (c) a starter cylinder that places the rod into the gripping device attached to the actuator hand, and (d) a drive motor that drives the band moving the rod through the coil.

Bond-integrity testing and sodium-level detection are both accomplished in a single operation. After the "B" magazine is removed from the bonding machine, it is transported to the test stand.

The orientation of the "B" magazine on the test station is fixed by means of an internal latching arrangement. By using the latching mechanism, the position of each pin undergoing the test is known. The test station is capable of indexing the magazine so that each of the 50 elements is brought to the test position. At the test position, the element is pushed up from its tube and through a test coil. A trace is made of the eddy current response versus the height of the element and the magazine position. The element is then lowered back into the magazine. Elements rejected after study of the eddy current trace are removed at a later stage. At the completion of this testing stage, the "B" magazine is transported to the spade-and-wire inspection jig.

Figure 38

EBR-II FUEL-ROD BOND TESTER(56)



Calculations have shown that voids larger than $\frac{1}{16}$ in. (peripheral width) by $\frac{3}{16}$ in. (axial length) will have deleterious effects on fuel-element heat transfer; the detection method must be capable of detecting accurately this size of void. Experimentation has shown(56) that voids of the order of $\frac{1}{32}$ in. in diameter can be successfully detected with the eddy current method. The effect of temperature of the sensing coil is important. Although the effect of self-heating may not be great, the approach that the bond test should be run with the sodium molten requires that the coil not react adversely to operation on fuel elements in the 100-200°C range. Initial measurements were made with a commercial system (Cyclograph) which operated at 90 kilocycles per second. For operations at high temperature (120-130°C), a frequency of 130 kilocycles per second was found to give results more accurate and less sensitive to temperature variation. In addition, tests at 30 kilocycles per second with a point probe rather than an encircling coil showed stability at the higher temperatures. At this writing, it was planned to employ the point probe coil in plant operations.

2.1.12.3 Sodium-level Testing(56,59)

Detection of the sodium level in a sealed fuel element has been difficult to accomplish. Initial attempts to discern the exact level were done with the use of an eddy current coil very similar to the one used for sodium void or bubble detection. It was found that the accuracy of this method and the acceptable tolerance for the sodium level were of the same order. Thus, the reliability of decisions based on this method was poor.

During the production of Core I, this method was so unreliable that an alternative method was introduced: X ray each fuel element and determine the actual level from the X-ray negative. This method was time-consuming and not adaptable to the remote process. The X-ray approach was used to keep Core I in production while another alternative method was developed.

A more sensitive, more reliable method was needed because the level to be measured was physically complex. When the sodium solidifies, an irregular surface is formed because of the meniscus and liquid-solid shrinkage. In many cases the sodium surface is not only dished, but is also higher at one side than at the other. It is believed that this surface tilt is caused by noncentering of a restrainer. The eddy current coil method of determination of sodium level is faced with a third complexity, namely, that near the sodium surface there is a discontinuity in the internal metallic content caused by a change in diameter of the restrainer.

The high-frequency eddy current methods reflect changes in the sodium skin but are also sensitive to the entire metallic content of the fuel element. As the element is raised through the coil, the presence of the fuel core is reflected. Just below the area of critical measurement, the top of the fuel pin is reached and the volume consists of solid sodium. Following the sodium volume, the end of the restrainer appears, and the restrainer has a varying diameter just at the region where the upper sodium surface should be. Therefore, the signal from the eddy current device becomes rather confused in the critical region. Recently, a twelve-coil, differential, point-probe, eddy current device has been used with very satisfactory results. This instrument has been calibrated against the X-ray data and shows good agreement. Additional development is required to produce a satisfactory instrument for remote production. Plans at this writing call for the use of a single coil of the point-probe differential type operating at 30 kilocycles per second for detection of both the void and sodium level (see Figure 37).

2.1.13 Fuel-element Inspection and Assembly into Reactor Subassemblies

The function of these operations is to combine completed fuel elements into a core subassembly that is acceptable for insertion into the reactor. The feed material consists of accepted fuel elements from the bond and sodium-level inspection station, as well as the externally produced components (blanket ends and hexagonal tubing). These materials are fabricated into a subassembly for the reactor. The actual assembly and final welding operations are preceded by an inspection and sizing operation. The final step of the remote refabrication operation is a tensile and straightness test on the completed subassembly.

2.1.13.1 Fuel-element Inspection

Although each pin-cladding jacket or can has been inspected before being sent into the glove box for sodium filling, a final inspection is performed to recheck the jacket for possible damage or alteration that might have occurred during fabrication. This step is not only an inspection procedure, but also a sizing or repair step. The sizing operation is pertinent specifically to the spade fitting at the end of the element. The spade, the exact size of which is critical to the assembly operation, is the section most susceptible to damage during the refabrication operation. Since the finished fuel element represents a high investment in remote operations and since many defects in the spade dimension can be corrected, a sizing jig has been incorporated into this inspection step. This jig will size the spade in the following ways:

a. correct slot width (0.022-0.026 in. wide);
b parallel legs to each other and to the element axis;
c. no upset or distortions on spade legs.

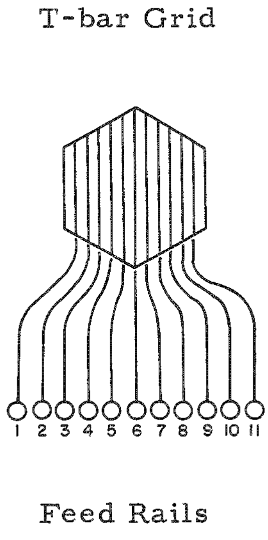
In addition, a visual inspection is performed to insure that elements with obvious dents, creases, or evidence of corrosion are not included in the subassembly.

A facet of the refining and refabrication process that has not been included in the plans to date is the cleanliness (radioactive contamination) of the fuel elements. The cladding jackets are cleaned before they enter the cell. Since they are new material, no problem of decontamination exists at this point. The sodium-filled jackets are transported into the argon cell. This cell, by design criteria will be an area of high contamination. The fuel elements, during the processes of fabrication, will become surface-contaminated. The major source of contamination of the air cell will probably be the contamination carried by the materials transferred out of the argon cell.

The question of cleanliness is relative; hence, contamination of the air cell is also an expected and tolerable condition. The contamination of the primary sodium system by fuel elements fabricated in the Fuel Cycle Facility would, in the long run, appear to be of little consequence. However, the situation bears checking in actual operations, and, if serious, facilities for cleaning fuel elements and subassemblies could be considered.

2.1.13.2 Assembly of Subassembly(60)

The remote assembly of 91 elements for a core subassembly (61 elements for a safety or control subassembly) into a hexagonal array, 2.3 in. across the flats, requires that control equipment be available which will insure the position of each of the components. In addition, a method of recording and indicating the correct loading sequence is required. Once the elements are in place, the outer jacket is mechanically forced over the elements and the jacket is welded to the end piece.



The initial operation is the insertion of the lower blanket piece into the assembly machine. At the upper end of the lower blanket piece is the T-bar grid. The grid is aligned to a feed rail system. At the entry end of each feed rail is a loading fixture (see sketch this page). The loading fixture serves two functions. It is, first, a "go, no-go" gauge to insure proper orientation of the spacer wire. In addition, the fixtures are electronically controlled to insure the proper sequence of loading. A programmer (located at the operating face of the air cell) indicates by a bright light the proper loading fixture indicated by the programmer; the other ten fixtures will not allow an element to pass. As the element passes through the loading fixture onto the feed rail, the bright light on the programmer is dimmed to indicate an occupied position.

As rows (normal to the loading direction) are filled, a thin holder blade is positioned across the front face of the row. Any misalignment of that row is indicated by failure of the holder blade to make contact with a switch and to light an indicator light on a console. The process is continued until the proper number of fuel elements have been loaded. Clamping fixtures hold the assembled elements in the proper hexagonal array while the holding blades are withdrawn. The clamping fixtures are not withdrawn until the hexagonal outer tube is started over the bundle. A forced-air cooling shroud which surrounds the rear two-thirds of the bundle is used to minimize the effect of self-heating.

Temperatures under conditions of self-heating of the refabricated subassembly have been calculated for coolant flow in the axial direction. The effectiveness of coolant flow normal to the fuel elements (as provided by the shroud) and for the possible heat rise with loss of coolant are cases which have not yet been considered. A series of experiments is underway which will indicate the heat rise with and without coolant.

To insure tight initial packing of the 91-element array, the outer hexagonal tube has been dimpled. These dimples are 20 mils deep and are positioned so that, in the finished array, they bear on the elements in the area where the helically wound spacing wire is not at the element-hex tube interface. The problem in the assembly operation is that, in sliding the hexagonal outer tube over the array, the dimpled (compressed) section must pass over areas where the helical wires cause a maximum width. To allow the hexagonal tube to slide into position, it is heat expanded just before it is forced over the element array.

It was pointed out above that a forced-cooling shroud was provided to prevent overheating of the 91-element array. The shroud serves in a dual capacity, as it is also designed to provide heating of the array. In the final stages of assembly, some plasticity of the fuel-element array, which aids in slipping the outer tube over the array, is provided by heating the elements to a condition at which the sodium is liquid. The liquid sodium allows some elastic deformation of the element bundle. In actual operation, either the heating or cooling capacities of the shroud may be called upon, depending upon the degree of self-heating. In the production of Core I, the element array was heated to approximately 250°C. The outer tube was heated to 750°C to 800°C, which provided the 500°C temperature differential necessary to permit easy assembly.

The force necessary to push the outer tube into position is measured and controlled. Forces used in the production of Core I varied between 125 and 300 lb. An arbitrary limit of 185 lb has been set as the maximum force to be used in the remote assembly. The unit is capable of exerting up to 2000 lb of force.

The outer-tube assembly (the outer tube is prewelded to the upper blanket and top piece before transfer into the air cell) seats on the shoulder of the lower end piece (a hexagonal flange) at the lower end of the lower blanket. Six condenser-discharge welding guns are positioned opposite the faces of the hexagonal flange. The welding guns are operated in sequence, and six welds are made between the outer tube and the lower flange.

When the outer tube assembly is pushed into place, the cooling provided by the shroud is no longer effective. The grapple used to grip the outer-tube assembly will be equipped to blow air through the assembly as soon as the outer tube is seated on the lower flange. This cooling grapple is the same as the one used to remove the subassembly from the interbuilding coffin.

The welding operation is the final manufacturing step in the refabrication process. This step is followed by a final test and inspection.

2.1.13.3 Final Test and Inspection

The subassembly is subjected to a 3000-lb tensile load, and the welds are checked for distortion and integrity. The subassembly is then subjected to a straightness check. The specifications on straightness allow for a deviation of 0.040 in. Checking for straightness will be done by lowering the finished subassembly through a "go, no-go" fixture which is the length of the subassembly and will allow a subassembly with a 40-mil bow to pass through under its own weight.

The question of whether or not subassemblies with excessive distortion will be straightened or refabricated has not been decided. Refabrication would be costly and time consuming; straightening would require equipment to discern the exact area of distortion and versatile equipment to counter the distortion.

2.1.13.4 Final Operations

Following the final inspection, the fuel subassembly is ready for return to the reactor. The cooling grapple (which is put in place when the outer tube is added to the unit) is used to place the subassembly in the transfer coffin for return to the reactor. At the reactor, the subassembly can be placed in the fuel-storage basket of the primary tank.

2.1.14 Scrap Remelting

Scrap metal composed of short pin castings, rejected pins, shards, and possibly heels from the injection-casting operation will be remelted into ingots for recycle (probably to the injection-casting operation, but possibly to the melt refining step). The purposes of remelting this material are to provide a separation of good metal from dross and to provide ingots suitable for recycle to the injection-casting operation.

No definite plans for special remelting equipment have been formulated; however, the regular melt refining equipment can be used. Either the regular melt refining crucibles of zirconia or special thoria-coated graphite crucibles could be used. If zirconia crucibles were used, the relative skull size would be less than in melt refining, since only melting and pouring without a liquation period would be required. If thoria-coated graphite crucibles were used, virtually no reaction with the crucible would occur to increase the amount of dross above that contained in the charge material.

Experimental work has been done on the remelting of injection-casting heels in the casting furnace. A special thoria-coated graphite remelt crucible with several holes in its bottom was placed above the regular crucible, and the heel pieces were charged into the upper crucible and melted. The molten metal ran through the holes into the lower crucible, leaving the cross with some metal behind as a skull or cake. This method of remelting has been used to separate pieces of broken injection-casting molds from heels.

2.1.15 Overall Operation and Flow Diagram

The detailed operations required to process fuel are shown in Figure 39. The major steps in the EBR-II fuel cycle were previously presented in Figures 1 and 2.

Figure 39

OVERALL PROCESS FLOWSHEET FOR EBR-II FUEL CYCLE

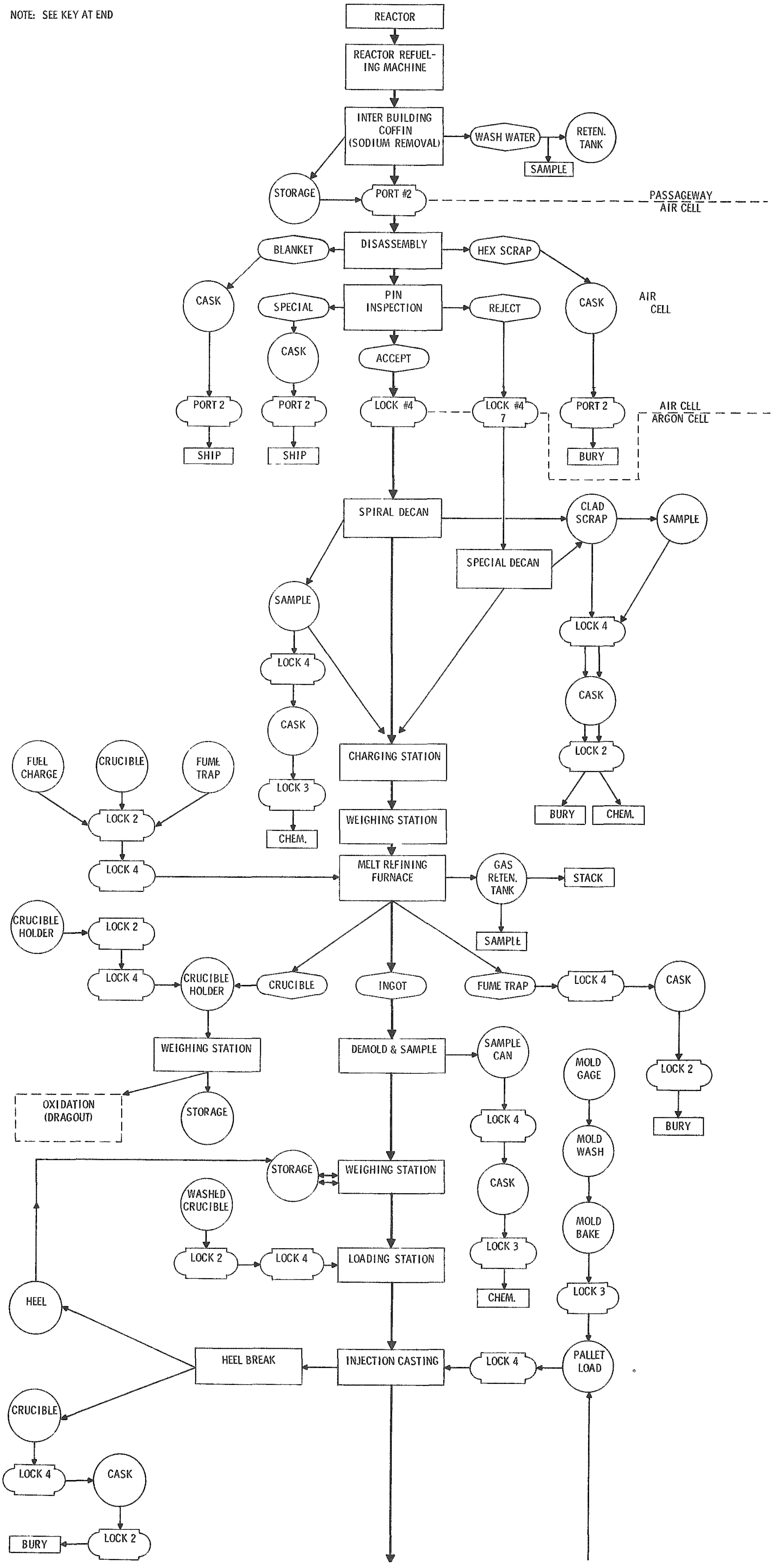

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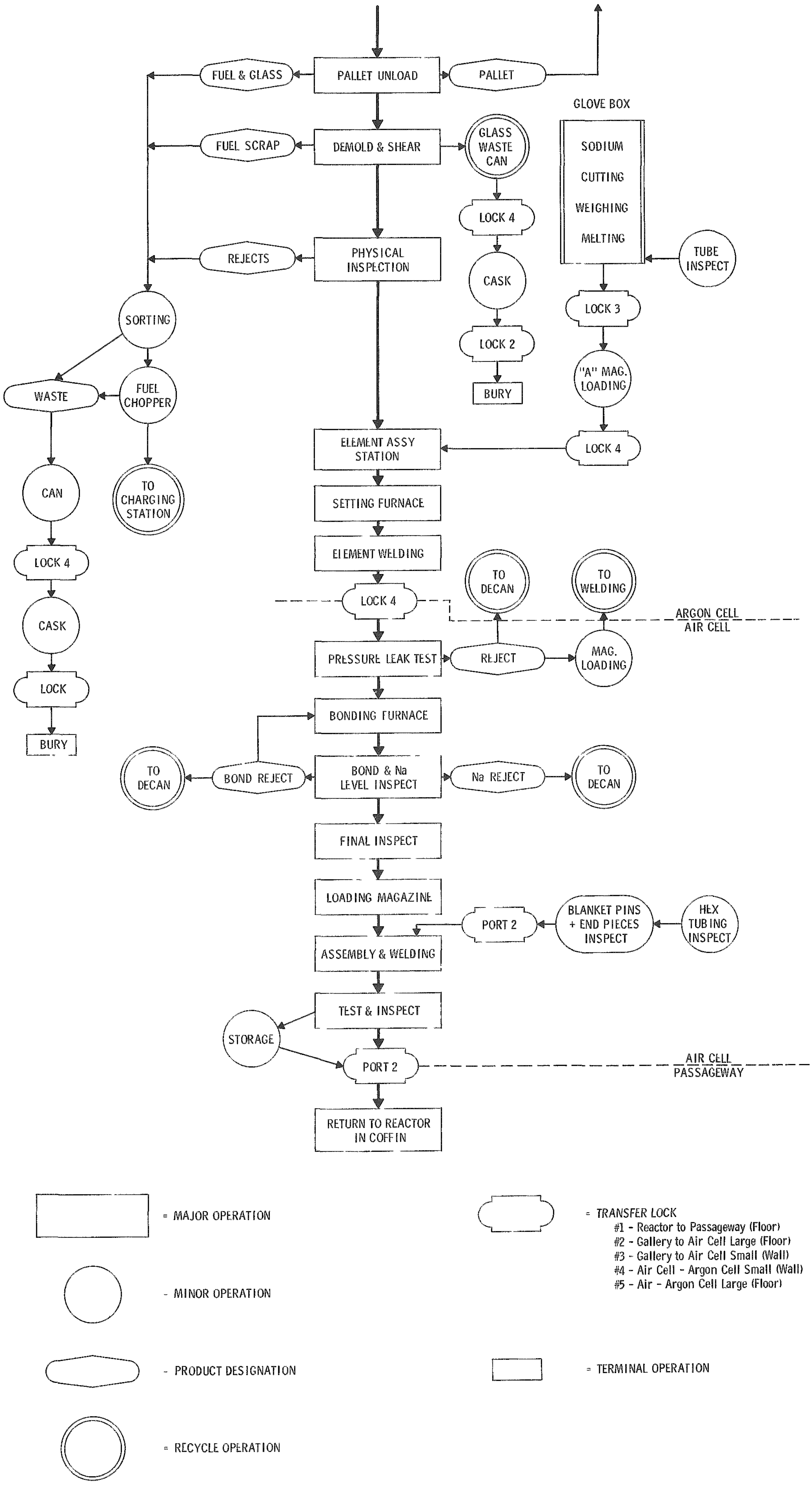
Figure 39 (Contd.)

OVERALL PROCESS FLOWSHEET FOR EBR-II FUEL CYCLE



2.2 Auxiliary and Alternative EBR-II Fuel Cycle Processes

In addition to melt refining, there are two other processes in the EBR-II fuel cycle. One of these is a process to recover the fissionable material remaining as skull material in the melt refining crucible. The residue or skull material remaining in a melt refining crucible constitutes about 10 percent of a melt refining charge. Under conditions of repetitive recycle of core material, the process should provide for removal of the relatively noble fission products (molybdenum, ruthenium, rhodium, and palladium) which are not removed in the melt refining process. In order to maintain equilibrium concentrations of these elements in repetitively recycled fuel, it would be necessary to process a small sidestream* of material for removal of the relatively noble fission products. A natural selection of material for processing in this sidestream is the skull remaining in the melt refining crucible. Therefore, two objectives need to be achieved in processing melt refining crucible residues: (1) the removal of noble fission product elements,** and (2) the recovery of the contained fissionable material. The process developed for this purpose is called the "Skull Reclamation Process."

The second additional process required in the EBR-II Fuel Cycle is a process for isolation of the plutonium bred in the EBR-II blanket of depleted uranium. The objective of this process is to isolate the plutonium or upgrade its concentration in uranium sufficiently so that it may be used for re-enrichment of EBR-II fuel cores. It is anticipated that plutonium will be allowed to "grow into" the blanket uranium to a concentration of about one percent before the blanket is discharged. The calculated minimum concentration of plutonium in uranium required for core re-enrichment is about 40 percent. Such plutonium-uranium alloy will be used, not in the first EBR-II core, which employs U^{235} as a fissionable material, but in subsequent cores which are intended to contain a large fraction of plutonium as the fissionable material. However, the blanket process will be demonstrated and made suitable for utilization with future plutonium cores.

The position of the skull reclamation (for reclaiming residual uranium from fission product slag) and blanket processes in the EBR-II fuel cycle is shown in Figure 2, page 16. These processes utilize molten metal and molten salt systems as processing media. The processes are, therefore, less compact than the melt refining process. However, there is considerable flexibility available in the choice of processing conditions by which various processing goals can be met.

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*In some reports, this sidestream material has been called "dragout" to indicate a deliberate removal of material from the main fuel cycle for processing to remove noble metals.

**Since alkaline and rare earth fission product elements are present in the skull material, having been removed from the bulk of the uranium in the melt refining process, these, too, must be removed from the skull material.

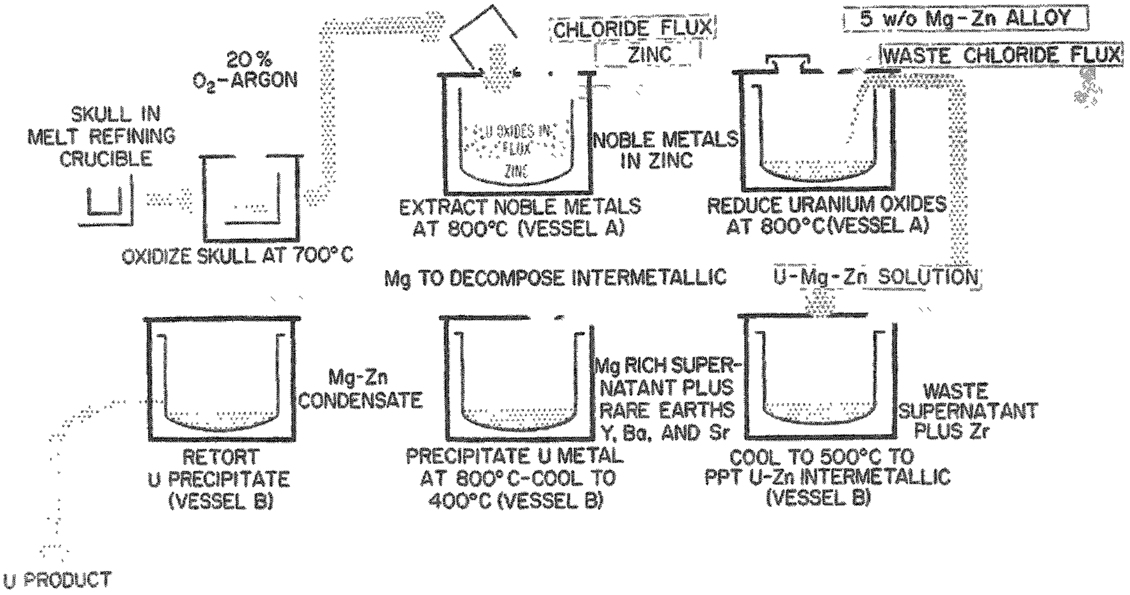
2.2.1 Skull Reclamation Process(36b,37c,39b,40,45b,48b,61)

2.2.1.1 Process Description

As mentioned above, the skull reclamation process complements the main melt refining process in the EBR-II fuel cycle. Its objectives are: (1) recovery of the uranium that remains as a lining in the melt refining crucible in a form known as skull material, and (2) separation of this uranium from accompanying fission products, thereby controlling the concentration of fission products in the recycled fuel. At the time of this writing, the skull reclamation process had become fairly well established in the form shown in Figure 40.

Figure 40

EBR-II LIQUID METAL PROCESS FOR RECLAMATION OF MELT REFINING SKULLS



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The skull material is first oxidized under controlled conditions (in a 20 percent oxygen-argon atmosphere at 700°C) to convert it to an oxide powder which is poured from the crucible. The oxides are suspended in a molten halide flux, and the noble metals (ruthenium, rhodium, molybdenum, and palladium) are removed by reduction with and extraction into zinc. The zinc is then removed and discarded. The flux-oxide mixture is then contacted with a dilute 5 percent magnesium-zinc alloy which reduces the uranium oxides and other fission product oxides. The liberated uranium and fission products dissolve in the zinc-magnesium phase. The flux containing the magnesium oxide reaction product is then discarded. The uranium-magnesium-zinc solution is transferred to a different crucible in which the uranium is precipitated as a uranium-zinc intermetallic compound (approximately U_2Zn_{23}) by cooling the solution to about 500°C. Removal of the supernatant solution serves to reduce the volume and also provides removal of zirconium (which is present as a fission product element and a contaminating element introduced as crucible fragments from the zirconium oxide melt refining crucible). The uranium-zinc intermetallic compound is then decomposed by the addition of magnesium, which extracts the zinc and precipitates uranium almost completely as metal. The rare earth fission products, as well as barium, strontium, and any plutonium present, are removed in the supernatant magnesium-rich zinc solution. The uranium precipitate (which is enveloped in the magnesium-zinc alloy) is then retorted to drive off the residual solvent metals and isolate the uranium product. Only a small amount of plutonium is generated during irradiation of the enriched uranium in the first core loading of EBR-II. Of this amount, the major portion (about 90 percent) is recycled from the melt refining process to the reactor. The remainder, which enters the skull recovery process, is not present in sufficient amounts to justify provisions for its recovery.

Typical fission product removals obtained in small-scale demonstration runs of this process were as follows:

Constituent	Percent Removal
Cerium	95-98
Zirconium	84-93
Molybdenum	87-92
Ruthenium	70-86
Palladium	99

Overall removals have ranged from 70 percent for ruthenium to 99 percent for palladium. For purposes of the EBR-II cycle, these removals are sufficient.

From analyses of waste streams, the overall uranium recovery in the skull reclamation process is expected to be above 95 percent. Since only a fraction of the total fuel material (about ten percent) is processed through the skull reclamation process, this recovery is adequate. A loss of five percent of the uranium processed via the skull reclamation process represents only 0.5 percent of the total fuel being processed. Losses expected in the waste streams from the skull reclamation process are shown in Table 9.

Table 9

EXPECTED URANIUM LOSSES IN THE SKULL RECLAMATION PROCESS

Step	% Uranium Loss Expected
Skull Oxidation	1 or less
Noble Metal Extraction	<0.1
Reduction	0.4
Intermetallic Compound Precipitation	0.3
Uranium Metal Precipitation	0.7
Total	$1.4 < \text{total} < 2.5$

2.2.1.2 Materials of Construction

Corrosion problems are particularly severe with the molten zinc alloy and fused salt systems used in the skull reclamation process. Only the refractory metals (such as tantalum, tungsten, and molybdenum), alloys of these, or very stable oxide ceramics (such as beryllia, alumina, or magnesia) possess sufficient corrosion resistance for consideration as container materials. After considerable testing of materials and development of suitable methods of fabrication, the following selection of materials has been made:

For containing systems of the noble metal-extraction and reduction steps	tungsten
For containing systems of the two uranium-precipitation steps	beryllia
For auxiliary purposes (agitators, transfer lines) in all systems	30 w/o tungsten-molybdenum alloy

Tungsten has proved to be essentially inert to the metal-salt systems used in the skull reclamation process. Corrosion tests under process conditions but at 850°C (50°C above the maximum processing temperature) have shown no detectable attack of tungsten after 500-hr exposures. Pressed-and-sintered tungsten crucibles have been apparently unaffected by process use for times of slightly over 1000 hr (accumulated use at the time of this writing). Tungsten is brittle at room temperature and must, therefore, be handled with care. However, at operating temperatures (500°C to 800°C) tungsten is ductile, since on heating it passes through a brittle-to-ductile transition at about 400°C. The least expensive and an eminently satisfactory method of preparing tungsten crucibles is by pressing and sintering. By this technique, it is possible to prepare crucibles large enough for full-scale operation, i.e., about 14 in. in diameter by 28 in. high. With the rapid developments in tungsten technology now accompanying the use of tungsten in space vehicles, it is possible that other methods of tungsten fabrication may eventually prove superior to pressing and sintering.

A ceramic material which is not wetted by the metal systems is required for the two uranium-precipitation steps so that removal of the uranium-product concentrate (enveloped in a magnesium-zinc alloy) can be achieved. Since tungsten is well wet by this alloy, removal of this uranium product concentrate from tungsten (or, for the same reason, from any other metal crucible) would be very difficult.

Beryllia, because of its high chemical stability and its good thermal conductivity (and hence good resistance to thermal shock), and because it is not wet by the metal systems, has received prime consideration as a container for use in the precipitation steps.* Methods for fabrication of large-size beryllia crucibles were not developed until recently when, in cooperation with the Brush Beryllium Co., a method of crucible preparation known as thixotropic casting was developed. In this preparation procedure, a thixotropic beryllia slurry is cast into the desired shape in a steel mold. The product is removed from the mold, dried, and fired. Crucibles prepared by this method are supposed to be relatively free of internal stresses. The thixotropic casting procedure appears applicable to the preparation of large-size crucibles, although none had yet been fabricated by this method at the time of this writing. Very promising results, however, have been obtained with intermediate-size crucibles (4 in. in OD by 9 in. high). Such crucibles have withstood 400 hr of accumulated process use without apparent deterioration. Essentially 100 percent removal of the precipitated uranium concentrate has been consistently achieved from such crucibles.

Because tungsten is difficult to machine, it was necessary to find a machinable alloy with suitable corrosion resistance for fabrication of auxiliary items, such as agitators and transfer tubes. Although less corrosion resistant than tungsten, a 30 percent tungsten-molybdenum alloy appears suitable for this purpose. Corrosion of the alloy occurs by slow leaching of the molybdenum. However, it is expected that component parts fabricated from this alloy will last well over 1000 hr. Because the items to be made from this alloy are easily replaceable, this lifetime is adequate.

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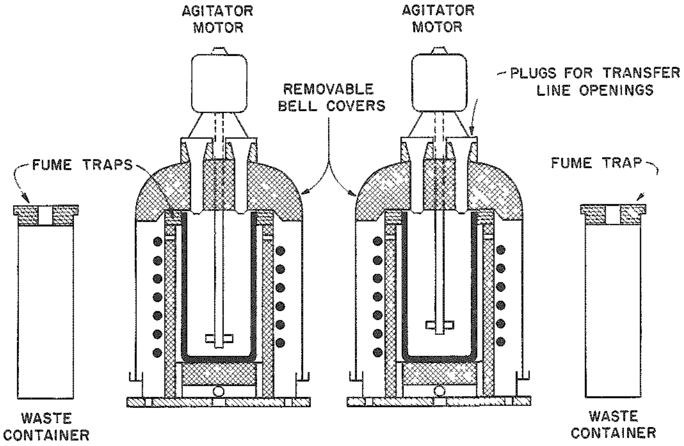
*It is possible that alumina or magnesia may also be satisfactory container materials.

2.2.1.3 Full-scale Equipment Design

The skull oxidation equipment was previously discussed on page 74. The design of the remainder of the full-scale equipment for the skull reclamation process had not been started at the time of this report. However, half-scale, integrated equipment had been designed and was under construction for process demonstration on a fairly large scale. It is thought that plant equipment will be very similar to the demonstration equipment. The design concept for this latter equipment is shown in Figure 41.

Figure 41

PREPLANT VERSION OF SKULL RECLAMATION PROCESS EQUIPMENT



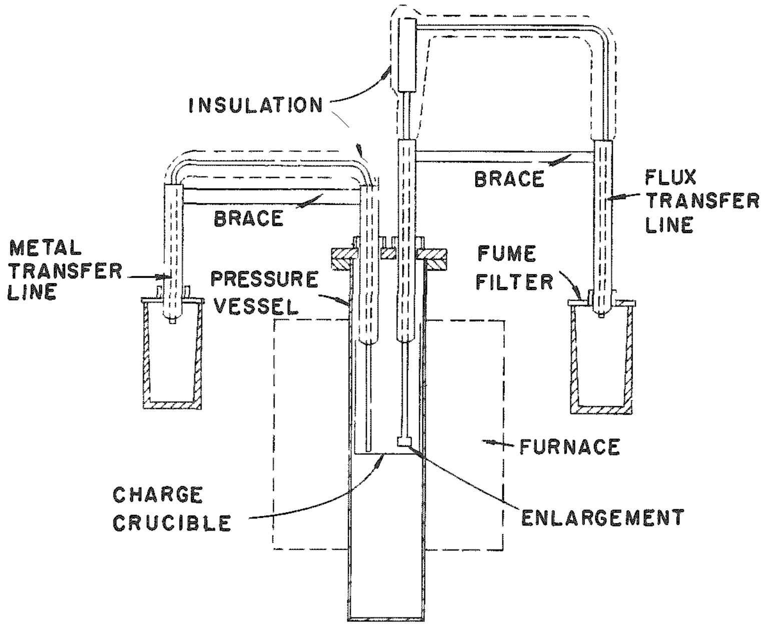
108-4763

The equipment consists essentially of two large, normally stationary, but easily replaceable, crucibles, each contained in a separate furnace. One of these, a pressed-and-sintered tungsten crucible (12 in. in OD by 19 in. high), serves for the first two steps of the process; the other, a beryllia crucible, of about the same size, serves for the two precipitation steps. It is expected that the product, a concentrate of uranium enveloped in a 50 percent magnesium-zinc alloy, will be retorted in a suitably modified melt refining furnace.

The crucibles are positioned centrally within metal bell jars (about 30 in. in diameter and about 40 in. high) which can be tightly sealed for maintenance of a high-purity inert atmosphere. Heating will be effected by means of graphite susceptors heated by uncoated induction coils. Solution removal for the necessary phase separations will be effected by pressuring the liquids out through dip tubes, a procedure known as pressure siphoning. Two suitably designed and located pressure-siphon tubes(95) can be operated together to provide automatic separation of a metal phase from a salt phase (see Figure 42). The liquids are transferred either to the adjacent crucible or to waste receivers.

Figure 42

TRANSFER TUBES FOR LIQUID METAL-SALT FLUX SEPARATIONS



108-5287

The major phase separation operations are:

1) Separation of molten zinc containing noble fission elements from the flux containing the skull oxide after noble metal leach in the first crucible. This separation is made by cooling to about 500°C, which freezes the flux but leaves the zinc molten. The zinc is then pressure siphoned from beneath the flux to a waste container.

2) Separation of molten zinc plus magnesium (containing the dissolved fuel from the molten flux) at about 800°C after the reduction in the first crucible. This separation is made by pressure-siphoning. The metal is directed to the second crucible, and the flux is directed to a waste container. The siphoning operations can be made individually or simultaneously by the automatic method.

3) Separation of molten zinc plus magnesium from the precipitated uranium-zinc intermetallic compound in the second crucible at about 500°C by pressure siphoning the zinc to a waste container.

4) Separation of molten magnesium plus zinc (containing fission products) from the precipitated uranium in the second crucible. The separation is made at about 500°C by pressure siphoning the magnesium plus zinc to a waste container.

5) Retorting the uranium cake to drive off the excess magnesium and zinc.

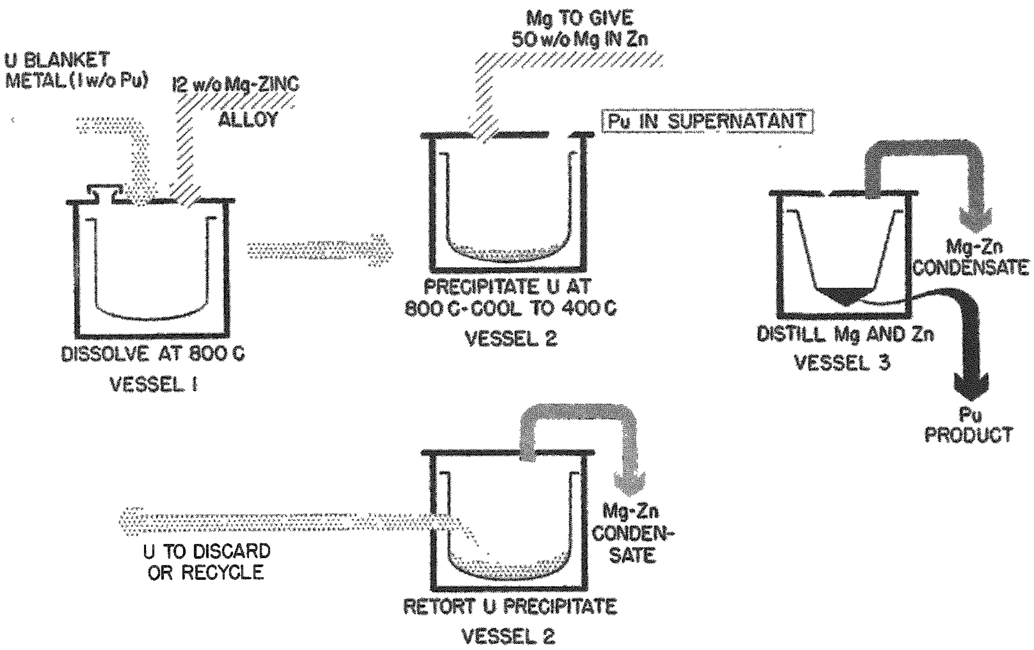
A recycle argon system can be used to accelerate the cooling of the crucibles for flux freezing and uranium precipitations.

2.2.2 Blanket Process(61,62)

The blanket process for isolation of plutonium from EBR-II uranium blanket material also utilizes liquid metal solvents. Separation of plutonium from uranium is based on the high solubility of plutonium in magnesium-rich solutions and the contrasting low solubility of uranium in such solutions. The process, outlined in Figure 43, involves dissolution of the blanket metal in a 12 w/o magnesium-zinc solution to a uranium concentration of about 14 percent, addition of magnesium to about a 50 percent concentration to precipitate the uranium away from the plutonium, separation of phases, and retorting the individual phases for recovery of plutonium and uranium.

Figure 43

EBR-II LIQUID METAL PROCESS FOR URANIUM BLANKET METAL



108-4740

In Table 10 are summarized the results of three demonstration runs of the blanket process. The blanket process shown in Figure 42 was modified slightly in that the uranium precipitate was washed once with a 50 percent magnesium-zinc alloy. The good separation of plutonium in high yield in these runs is very encouraging. The ratio of the weight of plutonium in the supernatant solutions to the sum of the weights of plutonium and uranium in these solutions [Pu/(U + Pu)] is about 0.6 in all runs. This substantially exceeds the minimum ratio of 0.4 required for enrichment of an EBR-II core. In these runs, about 70 to 80 percent of the supernatant product phase was removed in the phase separation operations. These rather low removals necessitated the use of a wash. Subsequent work on this phase separation has resulted in phase separations of over 95 percent, which would make possible elimination of the wash. This would result in a large increase in the plutonium separation factor (to around 250). It should be pointed out also that, since uranium continuously precipitates in the course of evaporation of the plutonium product solution, it is possible, by removal of a concentrated plutonium solution near the end of the evaporation, to separate nearly all of the uranium from the plutonium.

Table 10

SUMMARY OF BLANKET PROCESS DEMONSTRATION RUNS

(See blanket process in Figure 43)

Scale of Operation: 500 g of 1 percent Pu-U alloy

Run No.	Plutonium Recovery (%)	Product Pu/(Pu + U) Ratio	Separation Factor[a]
1	92	0.65	168
2	95	0.64	169
3	93	0.61	157

---

[a] Separation factor is defined as Pu/U ratio in product divided by Pu/U ratio in feed material.

The major problem remaining is the isolation of plutonium from the magnesium-zinc solution. Evaporation of the magnesium and zinc appears to be the most straightforward procedure.

The materials of construction will be the same as those employed in the skull reclamation process, i.e., tungsten in the uranium dissolver, and beryllia in the uranium precipitator. It is expected that the equipment will be very similar to that planned for the skull reclamation process (see Figure 41), except that, in addition, an evaporator will be needed for boildown of the plutonium product solution to a useful volume.

2.2.3 Other Liquid Metal Processes

Liquid metal processes are also being developed for the purification of other fuel materials, with major emphasis being given to future EBR-II core loadings, which are expected to contain plutonium as the fissionable material. Various actinide element compounds can also be prepared in liquid metal media. Uranium and plutonium carbides, silicides, and sulfides, for example, show considerable promise as high-performance reactor fuels, and it appears that pyrometallurgical techniques can be adapted to the preparation and processing of such fuels as well as metal fuels.

2.3 Physical Metallurgy of Core I, EBR-II Fuel (Uranium-Fissium Alloy)

2.3.1 Introduction

The purpose of this chapter is to review the physical-metallurgy data that has been developed with respect to the uranium-5 w/o fissium alloy fuel to be used in the first EBR-II core. Many facets of the physical metallurgy of uranium-fissium alloys have been investigated.(50,63,64,65) For example, Jelinek et al., have described work on the fabrication of EBR-II, Core I fuel pins.(50,51,66) As the equipment and procedures to be employed in the Fuel Cycle Facility are developed and established, this general work can be applied to the specific aspects of the process. At this writing, the complete designation of the parameters of the cycle is uncertain, as only after the actual startup of the fuel cycle will enough information be available on the process and the facility to allow a precise cycle to be designated.

2.3.2 General Metallurgy

The alloy referred to as fissium or fissium-type alloy (in this case, uranium-5 w/o fissium) may be defined as an alloy containing a near-equilibrium amount of fission products that would result from an infinite number of irradiation and pyrometallurgical reprocessing cycles (see Table 1, p. 19 for composition). The initial alloy composition has been adjusted to approach the calculated equilibrium composition and thereby reduce the magnitude of property changes inherent in the recycling of reactor fuel.

In general, the melt refining process removes the following elements nearly completely from the discharged fuel: krypton, xenon, rubidium, cesium, cadmium, strontium, barium, yttrium, the rare earths, tellurium, iodine, and bromine, either by volatilization or by selective oxidation and inclusion in an oxide dross. These elements comprise 67 w/o of the fission products generated per reactor pass. Fission products showing virtually no partition from the uranium metal in melt refining include zirconium, niobium, molybdenum, technetium, ruthenium, rhodium, palladium, silver, indium, tin, and antimony. These elements comprise 33 w/o of the fission products generated per reactor pass.

The estimated equilibrium concentrations of the elements that are not separated from the alloy on melt refining are shown in Table 11. While the concentrations shown in Table 11 are only approximations, they do show that, with the exception of technetium and zirconium, the alloy composition is not expected to show major variation from cycle to cycle.

Table 11

ESTIMATED EQUILIBRIUM COMPOSITIONS OF EBR-II FUEL ALLOYS

		Concentrations (w/o) in:		
Element	Fission Yield[a]	Initial Fuel Alloy[b]	Discharged Fuel Alloy	Melt-refined Alloy after 10% Dragout + ~10% Makeup
Zr	29.36	0.1	0.35	0.31[c]
Nb	1.58	0.01	0.023	0.02[c]
Mo	21.70	2.46	2.71	2.44
Tc	6.2	0	0.10	0.09[c]
Ru	15.73	1.96	2.12	1.91
Rh	2.81	0.27	0.31	0.28
Pd	1.61	0.19	0.21	0.19
Ag	0.10	-	-	-
In	0.04	-	-	-
Sn	0.34	-	-	-
Sb	0.22	-	-	-

---

[a] Fission yield is in atoms per 100 atoms of U^{235} fissioned after 135 days of irradiation and 15 days of cooling.

[b] This alloy is a nominal alloy composition - a calculated equilibrium composition for about seven percent "dragout" which was designed to give a total alloying element concentration of 5 w/o. It differs from that given in Table 1, p. 19, which is the actual fuel composition. The slight changes occurring in molybdenum, ruthenium, and rhodium concentrations in the last column reflect the use of a 10 percent "dragout" rather than a seven percent "dragout."

[c] Not equilibrium concentrations. The concentration changes are those which occur in the first fuel pass through the reactor. However, because of the 35-day half-life of niobium, its equilibrium concentration will remain in the vicinity of 0.01 percent.

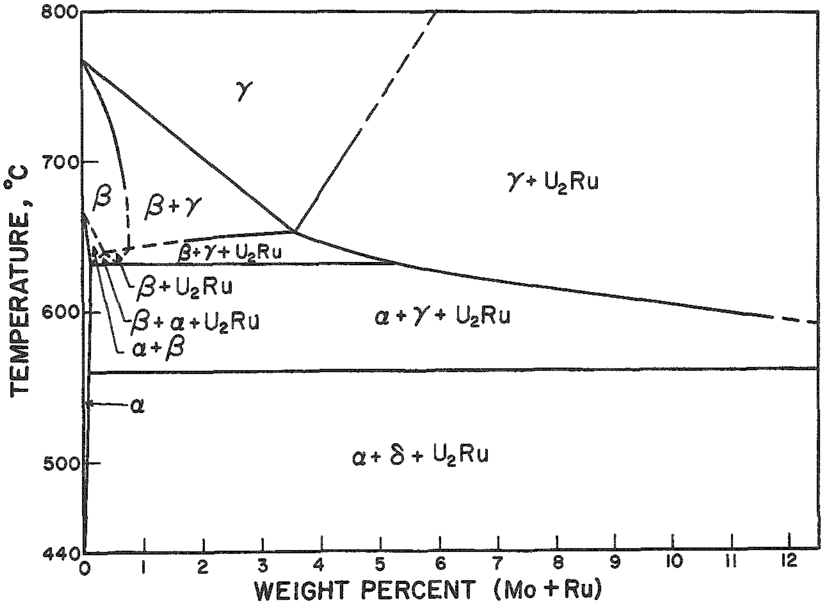
Of the eleven metals which tend to build up in the recycled fissium alloy, technetium is the one whose effect is least known. Since no stable isotope of technetium exists, little experimental work has been done with it that would be pertinent to this alloy. The chemical behavior of technetium resembles that of rhenium, and its crystal structure is the same as that of rhenium and ruthenium. The effect of buildup of this element in the alloy will be of major interest. In studies of the alloy, the amounts of molybdenum and ruthenium were increased as a substitute for technetium.(67)

Calculations indicate that about 0.25 w/o of zirconium will be generated in each pass of EBR-II fuel. If a relatively minor removal of zirconium in melt refining is effected (about 5-10 w/o is expected), the fuel resulting from 10 passes (with make-up uranium added that contains no zirconium) will contain approximately 1.7 w/o zirconium in addition to 5 w/o fissium. This alloy could be more reasonably referred to as U-5Fs-1.7 Zr (concentrations in weight percent). In the light of this probable increase in the zirconium content, work has been done on an alloy whose composition is U-5 Fs-2.25 Zr.(68)

In studies of a family of uranium-fissium alloys, Zegler and Nevitt(67,68) found that the phase relationships parallel closely those in the dominant U-Mo-Ru ternary system which has been studied by Dwight.(69) Figure 44 shows a vertical section of this ternary system in which the weight ratio of molybdenum to ruthenium is 1 to 0.8, approximately the ratio of the two elements in the uranium-fissium alloy.

Figure 44

VERTICAL SECTION THROUGH THE URANIUM-RICH CORNER OF THE U-Mo-Ru TERNARY SYSTEM AT THE Mo:Ru RATIO OF 1:0.8(69)



ANL Met. Micro 30162

The parent metal in this alloy, uranium, exhibits anisotropic behavior, particularly in thermal expansion. The metal is allotropic, exhibiting an alpha phase (orthorhombic) stable up to 668°C, a beta phase (tetragonal) stable from 668°C to 774°C, and the gamma phase (body-centered cubic) stable above 774°C. In the alpha phase, the anisotropic thermal-expansion properties* have given rise to permanent growth or distortion of uranium fuel sections upon thermal cycling. In the beta phase, the coefficients of thermal expansion show anisotropic behavior but of less severity, as the expansion in two crystallographic directions is equal. In the cubic gamma phase, no anisotropy is present.

One of the major reasons for alloying reactor fuel is to inhibit the transformation from the gamma to the alpha phase, so as to make possible the retention of the gamma phase at reactor operating temperatures. Of equal importance are the advantageous changes in physical properties of the uranium alloys. It should be added that where pure (or nearly pure) uranium is used, a heat treatment which produces a random grain orientation is used, thereby minimizing the effects of anisotropy. The problem is that most fabrication methods produce a preferred orientation in the grain structure which cannot be fully randomized by heat treatment. In the uranium-fissium (U-Fs) alloy for EBR-II, the fuel rods are cast to exact diameter and sheared to length. This method of fabrication does not introduce a preferred grain orientation so that dimensional changes on thermal cycling are expected to be minimized.

As mentioned earlier, the phase-transformation relationships of the uranium-fissium alloys are similar to those of the dominant uranium-molybdenum-ruthenium ternary alloy. A comparison of the phase tranformations is shown in Table 12. The phase relationships of the uranium-fissium alloys, in addition to the thermal-expansion phenomena, are important to the fuel-cycle process by reason of their effects on hardness, density, and radiation resistance of the alloys.

Table 12

COMPARISON OF PHASE TRANSFORMATIONS IN VARIOUS URANIUM ALLOYS(67,68)

	Temperature (°C)		
	U-Mo-Ru [5 w/o (Mo + Ru) Ratio 1.0:0.8]	U-5 w/o Fs	U-5 w/o Fs-2.25 w/o Zr
Gamma $\rightleftarrows$ Gamma + U_2Ru	~750 (1380°F)	725 (1337°F)	
Gamma + U_2Ru $\rightleftarrows$ Gamma + Alpha + U_2Ru	~625 (1157°F)	642 (1188°F)	
Gamma + Alpha + U_2Ru $\rightleftarrows$ Alpha + Delta + U_2Ru	~560 (1040°F)	552 (1025°F)	
Gamma $\rightleftarrows$ Gamma + ZrRu			Above 975 (1787°F)
Gamma + ZrRu $\rightarrow$ Gamma + Alpha + ZrRu			662 (1224°F)
Gamma + Alpha + ZrRu $\rightarrow$ Alpha + Delta + ZrRu			552 (1025°F)

---

* Alpha uranium is orthorhombic. Its mean thermal expansion coefficients between 25 and 125°C in the three principal crystallographic directions are: $a_0 = 21.7 \times 10^{-6}$, $b_0 = -1.5 \times 10^{-6}$, and $c_0 = 23.2 \times 10^{-6}$ in./in./°C.

Although the addition of alloying agents does reduce the temperature at which the gamma-to-alpha transformation takes place, a more practical aspect of the alloy addition in fuel preparation is that of making the transformation rate sluggish enough so that a metastable gamma phase can be obtained by quenching.

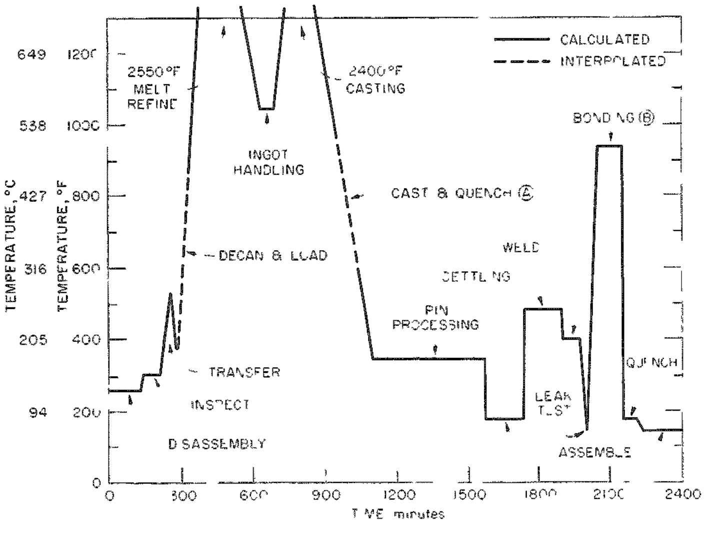
2.3.3 Factors Relevant to Pin Production

2.3.3.1 Temperature and Phase Transformations

It is important to the refabrication operations that the alloy contain retained gamma phase. This requirement is the result of the high hardness values and the density increase which occur as a result of the transformation from the gamma to the alpha phase. Experience in the fabrication of the initial core loading for EBR-II showed rather conclusively that an increase in hardness of from about 200 V.H.N (Vickers Hardness Number) for the gamma-retained alloy to about 500 V.H.N. for the alpha-transformed material made fuel-pin shearing a difficult, if not impractical, operation. In addition, the attainment of proper sodium levels in canned fuel elements is complicated by dimensional changes which accompany phase changes that occur after the casting step. Methods have been found to control the transformations in the production of the initial core loading. In actual fuel-refabrication operations, the effect of self-heating may have a pronounced effect on maintaining the required phase relationships. A plot of the expected temperatures versus time (or operation during fuel reprocessing) is shown in Figure 45. Two areas of possible importance in the time-temperature phase relationship have been designated on Figure 45 as A and B.

Figure 45

EXPECTED FUEL TEMPERATURES DURING EBR-II PROCESSING CYCLE (135 Days Irradiation to 2 a/o Burnup, Followed by 15 Days of Cooling)



2.3.3.1.1 Casting and Quenching

In the discussion of sodium level in fuel elements, (see p. 105) the importance of blast quenching was discussed. It was mentioned that the retention of pure gamma phase in cast pins is improbable, but that the adjusted cooling rate provided by the blast quench did appear to equalize the percent of retained gamma phase throughout the batches of 120 to 160 pins. In addition to effecting an equalization of the percent of gamma phase retained, the blast quench technique also helps ensure that enough of the gamma phase is retained to permit the shearing operation to proceed without difficulty.

The uranium-5 percent fission alloy as cast in the laboratory has a hardness of approximately 200-225 V.H.N. The alloy given a gamma quench (825°C for 66 hr and water quenched) has a hardness of 150-200 V.H.N. The alloy as cast in the prototype injection-casting furnace without the benefit of blast quenching has a hardness of approximately 267-312 V.H.N. (63-66 Rockwell A).(70) Indications from the production of Core I(71) are that if enough retained gamma can be maintained to keep the fuel hardness in the range of 350-450 V.H.N., the end shearing operation, as now designed, will operate adequately. The fully alpha-transformed material (alpha + delta + U_2Ru) has a hardness of about 550 V.H.N. and is not amenable to the present shearing methods. Losses of the order of 30 percent due to improper shearing and breakage have resulted when alpha-phase fuel was processed.

2.3.3.1.2 Bonding

An examination of Table 13 shows that, for the temperatures used in the bonding operation (500-550°C), the alloy is fully transformed to alpha before the 3-hr bonding period has elapsed. A report(56) dealing with the methods used during Core I production and with the proposed "impact" method of bonding states that maximum bonding efficiency was reached at a 500°C bonding temperature. Lowering the temperature decreased the bonding efficiency, although it was believed that, if decreased temperatures were required, an acceptable lower limit would be 400-450°C. The hardness data presented in Table 13 indicate that, if the alloy is held at 375°C for 3 hr and if the as-cast rather than the gamma-quenched alloy is used, the alpha transformation will go to completion. This indicates that the 400-450°C bonding range would not be low enough to be effective in retaining a gamma-phase alloy.

Related to this discussion of the phase-transformation effect of the bonding operation (for here the concern is with the fuel sent to the reactor) is the fact that the alloy, after completion of the subassembly, is placed in a 700°F (370°C) sodium environment in the fuel-storage rack. Although any individual element may have a long (days) or short (hours) stay in the fuel-storage rack, it is almost certain that the alloy will experience the 3 to 5 hr at 375°C necessary for alpha tranformation.

Table 13

PHASE-TRANSFORMATION RATES OF URANIUM-FISSIUM ALLOYS(67) AS INDICATED BY CHANGES IN VICKERS HARDNESS NUMBERS[a]

Heat Treatment		Vickers Hardness Number	
Temp. (°C)	Time	U-5% Fs (As Cast - 214)	U-5% Fs - 2.25% Zr (As Cast - 375)
825	66 hr	156 - 219	261 - 289
500[b]	5 min	455	561
	15 min	549	546
	25 min	542	536
	45 min	546	536
	1 hr	552	553
375	5 min	252	452
	15 min	256	481
	25 min	261	493
	45 min	268	506
	1 hr	272	521
	5 hr	553	549
300[c]	1 hr	200	
	3 hr	230	
	5 hr	240	
	7 hr	260	
	10 hr	300	
200[c]	1 hr	190	
	10 hr	210	
	30 hr	220	
	70 hr	260	

---

[a] A uranium-5 percent fissium alloy has a Vickers Hardness Number of 150-200 when in the gamma phase and of 550 when in the alpha phase. Intermediate numbers indicate partial transformations from the gamma to the alpha phase.

[b] Heat treatments at 500, 375, 300, and 200°C were preceded by heat treatment for 66 hr at 825°C and water quenching.

[c] Data at 200-300°C is from BMI-1123(63) and is for an alloy of approximately the uranium-5 percent fissium composition.

At reactor operating temperatures (central fuel temperature of 1150-1200°F), the alloy will transform to a predominantly gamma and alpha + U_2Ru structure. Those elements whose temperature exceeds 642°C will transform to a gamma + U_2Ru structure. There is little evidence that any irradiation-induced phase reversals, such as are seen in the uranium-molybdenum(64,72,73) alloys, are exhibited by this alloy.

The specific effect of phase cycling under irradiation is an area worthy of further investigation, since here is a possible source of deleterious radiation effects. In the initial core loading, 105 elements distributed among 15 subassemblies have been given special preparation and will serve as a basis for an analytical study of the fuel at operating temperatures experienced under actual reactor conditions. Plans for this experiment include the removal of a portion of the material at less than maximum burnup, so that an indication of the effects of radiation over a burnup range can be learned.

2.3.3.2 Effects of Additional Fission Product Alloying Occurring during Fuel Burnup and Recycle

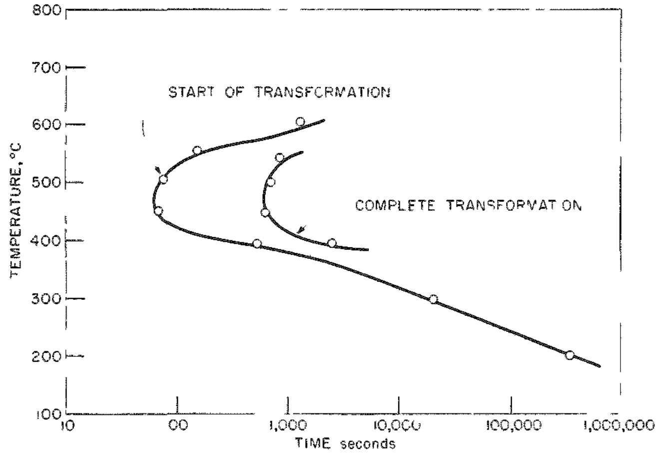
2.3.3.2.1 Zirconium

Data in Table 11 indicate that the uranium-5 w/o fissium alloy has an equilibrium composition except with respect to technetium and zirconium. It is known that zirconium is not removed effectively during the present melt refining process.(32) The zirconium content can be reduced by the addition of carbon, however, since, under melt refining conditions, zirconium carbide is more stable than uranium carbide. The carbon drossing causes about a 2 percent decrease in yield and increases the carbon content of the alloy to 350-500 ppm. However, because of difficulties in the mechanics of remotely adding controlled amounts of carbon to the alloy, the carbide-drossing procedure has not been included in the process.

The consequence of not reducing the zirconium content by carbide drossing is an increase of zirconium content in each successive reactor cycle. Because of the expected zirconium concentration increase, an alloy of the composition of uranium-5 percent fissium-2.25 percent zirconium was investigated.(67) As is seen in Table 13, the alloy is initially harder (both as cast and annealed) than uranium-5-percent fissium, and retention of the gamma phase by quenching will be much more difficult. At temperatures near 375°C and at times of the order of 0.1 hr, the alloy reaches the hardness range which causes shearing difficulties. While there is no experimental evidence about alloys with zirconium concentrations between the uranium-5 percent fissium (zirconium concentration is 0.1 percent) and uranium-5 percent fissium-2.25 percent zirconium, it may be supposed that as the zirconium content increases, the sluggishness of the gamma-to-alpha transformation disappears. Figure 46 shows the time-temperature-transformation (TTT) diagram for an approximate uranium-5 percent fissium alloy.(63) The effect of the zirconium increase will be to push the knees of the curves to the left and make retention of the gamma phase more difficult.

Figure 46

TIME-TEMPERATURE RELATIONSHIPS FOR THE BEGINNING AND END OF THE HARDNESS INCREASE IN AN ISOTHERMALLY TRANSFORMED FISSIUM ALLOY (BMI-1123)(63)



2.3.3.2.2 Technetium

Of interest in process metallurgy will be the effect of technetium on the alloy properties. This element is produced during fuel burnup to the extent of 2 a/o at the rate of approximately 0.1 w/o per cycle. Indications are that no effective removal of technetium will occur in the melt refining process.(34) It has been estimated(74) that, after many cycles, the alloy composition will contain about one weight percent technetium. As was mentioned earlier, because alloy studies with technetium have to be done with radioactive material, little information is available on the alloying properties of this element. No information is available on the alloying effect of technetium in the fissium alloys.

2.3.3.3 Effects of Contaminants on Fuel Alloy Properties

There are many possible contaminants of the alloy which can be introduced in fuel-reprocessing operations. Metallurgically, the closed-cycle aspects are an innovation, since, in general, metals are subject to only one alloying cycle and the dilution percentages in scrap recycle normally are large. In this particular case, contaminants picked up during the remelting and refabrication cycles, unless removed in the melt refining stage, can build up in the alloy. Of the materials that can be considered contaminants in the alloy, the following (and their sources) are discussed:

Contaminants	Source
Thoria	Thoria mold wash
Iron, Nickel, Chromium	Cladding material
Silicon	Vycor molds

2.3.3.3.1 Thoria Wash

Each mold is coated internally with approximately $\frac{1}{2}$ g of thoria wash (98 percent thoria-2 percent kaolin). If as much as one-half of the wash were to stay with the pin (an unlikely possibility), the contamination from this source would amount to 22 g of thoria per 6-kg batch of alloy. While no measurements were taken during the production of Core I, visual observations indicate that no more than trace amounts of the wash adheres to the cast pin. The thoria is not expected to be deleterious to the melt refining process, and it is expected that it will be removed by melt refining.

2.3.3.3.2 Stainless Steel Contaminants (Iron, Nickel, and Chromium)

The possibility exists that chips, dust, or small sections of stainless cladding can get into the alloy from the decanning operation. Iron, chromium, and nickel will not be removed in the melt refining step, and the effect of small amounts of these elements on the alloy are not known.

2.3.3.3.3 Silicon

Of the possible contaminants, it is likely that silicon will be in the alloy. Two major avenues of entry exist for silicon. The first is dust and chips from the mold crushing operation. The second and more probable avenue of entry for silicon is by means of sections of Vycor mold which become part of the melt. As noted in the section on injection casting (p. 76), a mold can break due to thermal shock and drop into the heel. Extraction of the mold sections left in the heel can be done by remelting the alloy in a perforated crucible. During the time the injection-casting heel is molten and during remelting, however, reaction of the melt with Vycor at the uncoated mold surfaces will occur. Hampson(75) has shown that silicon dissolved in the alloy is not removed by melt refining.

Zegler,(76) in an investigation of additions of from 0.25 to 2.00 w/o of silicon, aluminum, titanium, and yttrium to a uranium-5 percent fissium-2.25 percent zirconium alloy, showed that the addition of silicon to the alloy in amounts of less than 2 percent has a beneficial effect on the phase-transformation properties of that alloy. Although an increase in silicon content did cause an increase in the as-cast hardness (0.25 percent silicon gave 239 V.H.N.; 0.5 percent gave 257 V.H.N.; and 1.0 percent gave 309 V.H.N.), the major effect of the silicon was to increase the retained gamma in the as-cast alloy and thus lower the final hardness. As-cast uranium-5 w/o fissium-2.25 w/o zirconium plus 0.25 w/o silicon had a V.H.N. of 239. Between 1 and 2 percent silicon, the beneficial effects gradually disappear, and above 2 percent, the as-cast alloy is very hard (639 V.H.N.). It appears that for the purposes of Fuel Cycle Facility operations, the addition of small amounts of silicon to the alloy will not be detrimental.

2.3.4 Summary

The alloy used in the initial loading of the reactor is a synthetic combination of uranium and noble metal fission product elements. The use of the uranium-5 w/o fissium alloy is a technique for reducing property changes in the alloy as buildup of fission products occurs. Equilibrium concentrations of all elements (except zirconium and technetium) not removed by the melt refining process have been used in constructing the uranium-5 w/o fissium alloy.

The kinetics of phase transformation of the alloys are important to the process because of hardness and density changes that accompany the phase changes. Hardness is important in the shearing and handling operations, while the difference in density of the two major uranium phases (alpha and gamma) is paralleled by a difference in fuel-pin volume, which in turn affects sodium level in the fuel element.

The increase in zirconium content occurring during successive fuel cycles has been studied at high zirconium contents (uranium-5 w/o fissium-2.25 w/o zirconium). Because of the increased hardness of the alloy caused by the pressure of zirconium, difficulties in shearing pins to length can be expected with high-zirconium alloys. Although the increase in technetium concentration will be less than the increase in zirconium concentration, the effect will be of interest, as little information exists on the effect of this alloying agent.

An additional area of metallurgical interest will be the contamination of the alloy as it is recycled. Major possible contaminants are thoria and silicon (from the molds), and iron, chromium, and nickel (from the cladding material). There is a possibility that these elements, which are not removed in the refining operations, will build up to concentrations that will affect the refabrication and use of the alloy. Since little information is presently available on the effect of these elements in the alloy, actual operations will be required to indicate the magnitude of any resulting problems.

2.4 Fuel Movements and Storage

2.4.1 Fuel Cycle Flow Diagram

2.4.1.1 Fuel Inventories

In the flow of fuel through the Fuel Cycle Facility, inventories or accumulations of fuel will occur at certain points in the cycle. This accumulation is due to the time required to move and process the fuel and because of minor mismatches in the processing operations.

It is difficult to specify an expected accumulation of fuel in the process. Initially, work will most probably be done with one or two assembly equivalents of fuel in the argon cell. When efficient operation is achieved, it has been estimated that six assembly equivalents of fuel will be necessary to keep the processing equipment in operation. One can estimate that the fuel accumulation or in-cell inventory will then be between 6 and 12 assembly equivalents.

The storage of subassemblies, fuel and blanket elements, and uranium is not expected to create unusually difficult problems except possibly for exposed fuel which could react with nitrogen present at a few percent in the argon atmosphere. This possibility would be increased by extended storage of exposed fuel.

The storage of irradiated blanket elements is not expected to be a problem since self-heating of blanket elements from core subassemblies would be less than one watt per blanket element (for one pass through the reactor at 2-a/o fuel burnup).

The problem of nitridation of exposed fuel in the argon cell may require special provisions in the case of ingots, injection-casting shards, pin rejects, and heels awaiting remelting. Special argon cooling or a storage container with a very low nitrogen concentration in its atmosphere may be required. Nitridation may also be a problem in the handling of pins between decanning and melt refining and between injection casting and recanning (see discussion of nitridation problems, p. 182).

The storage of melt refining skulls in the crucibles or of the skull oxide (if an oxidation step is used) will present special problems mainly because of the large number which may accumulate before reprocessing arrangements are made. If only decanned pins are melt refined, and if injection-casting shards, rejects, and heels are remelted separately, a minimum of about ten percent of the fuel discharged from the reactor will end up as skulls or as skull oxide. For full reactor power and 2 percent burnup, this would result in about one 600-g skull or equivalent in skull oxide every two days (this assumes a 90 percent melt refining pour).

There are seventeen 12-in.-diameter by 10-ft-deep, water-cooled storage pits in the argon cell and ten (of the same size) in the air cell. If eleven of these pits were used for skull or skull oxide storage, about 100 skulls in crucibles or the equivalent in skull oxide could be stored. This would be about one year's production. The skulls could be added to the pit storage racks in a staggered manner so that, as the self-heating decayed, more skulls could be added to a given pit. The problem of skull storage is discussed further in Section 2.4.6, p. 147 of this chapter.

2.4.1.2 Flow Diagrams

A number of types of flow cycles of the fuel from the disassembled fuel pins back to refabricated fuel pins can be visualized. In general, the fuel-flow cycles are governed by the following factors, which were identified in test work with unirradiated fuel alloy:

1) Melt refining charge weight can vary; the maximum charge will be about 11 kg. At maximum charge, about ten percent of the melt remains as skull. Small charges have a larger percentage of the melt as skull.

2) In injection casting, only about 45 percent of the charge is converted to pins suitable for canning. Of the charge, about 23 percent is heel (which normally is recharged to the injection casting furnace without remelting), 31 percent is processed as shards and rejects, and 1 percent is unusable residue. Injection-casting charge weight can vary, but charges of about $12\frac{1}{2}$ kg have been used in the experimental work. The number of Vycor pin molds varies with the charge weight and is 120 for $12\frac{1}{2}$ kg.

3) Enriched uranium must be added to compensate for burnup.

4) Uranium must be added as metal or as fuel alloy to compensate for processing losses.

5) Mixing in the injection-casting crucible is not as good as in the melt refining crucible due to shorter liquation time.

Figures 47 through 51 show possible flow diagrams. Four of the flow diagrams are based on starting with 91 fuel elements (the number in one core subassembly). One of the flow diagrams is based on 85 fuel elements (the average number of elements in core, control, and safety subassemblies).

Since an inventory of disassembled fuel elements can be maintained without serious problems, the number of elements to be decanned for a melt refining batch can be varied to suit best operating

Figure 47

ALTERNATIVE FUEL MATERIAL FLOW DIAGRAM FOR EBR-II FUEL CYCLE - VERSION 1 (Makeup Fuel to Replace Losses Added as New Fuel Elements - Scrap Remelted Separately)

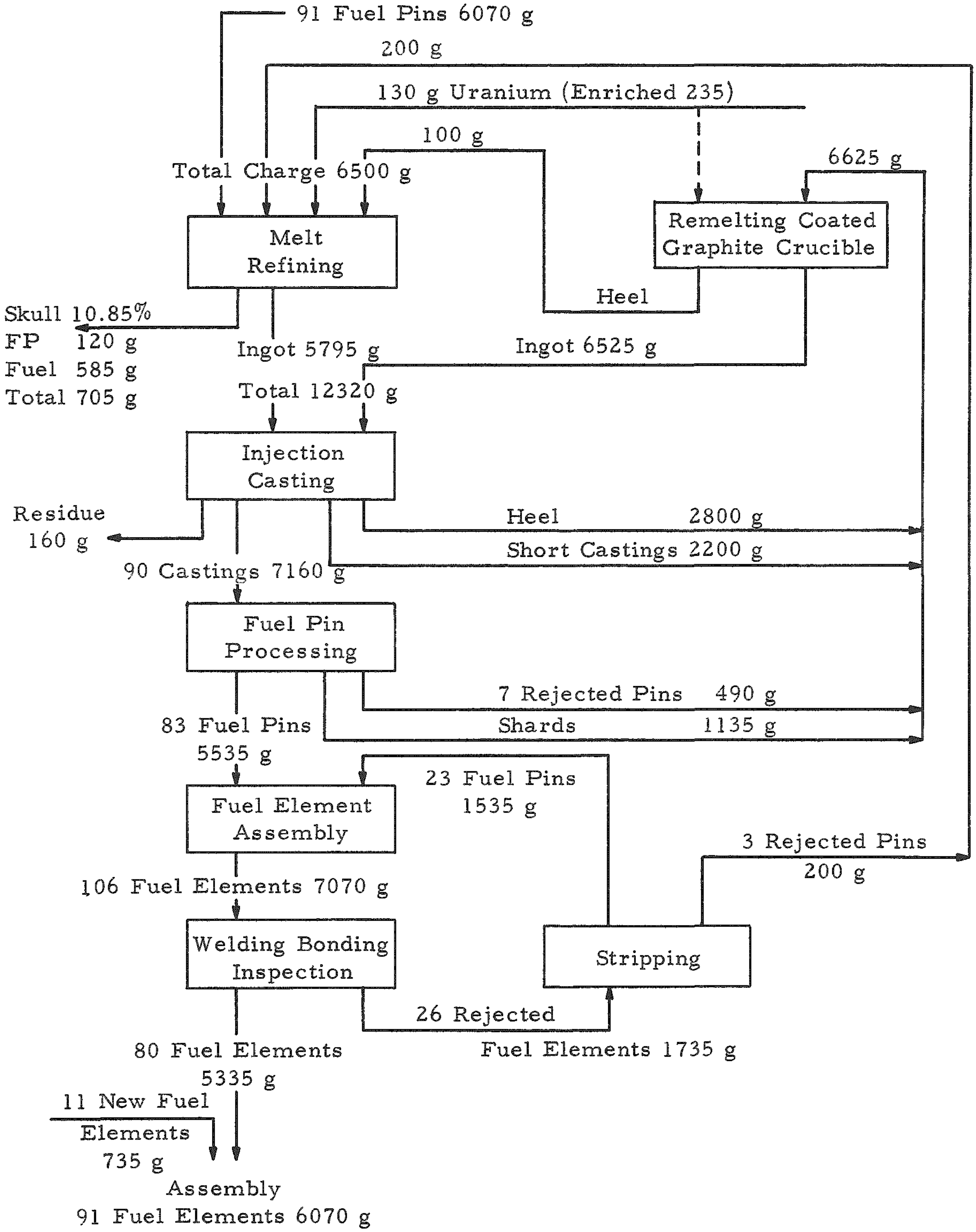


Figure 48

ALTERNATIVE FUEL MATERIAL FLOW DIAGRAM FOR EBR-II FUEL CYCLE - VERSION 2 (Makeup Fuel to Replace Losses Added in Scrap Remelt Operation)

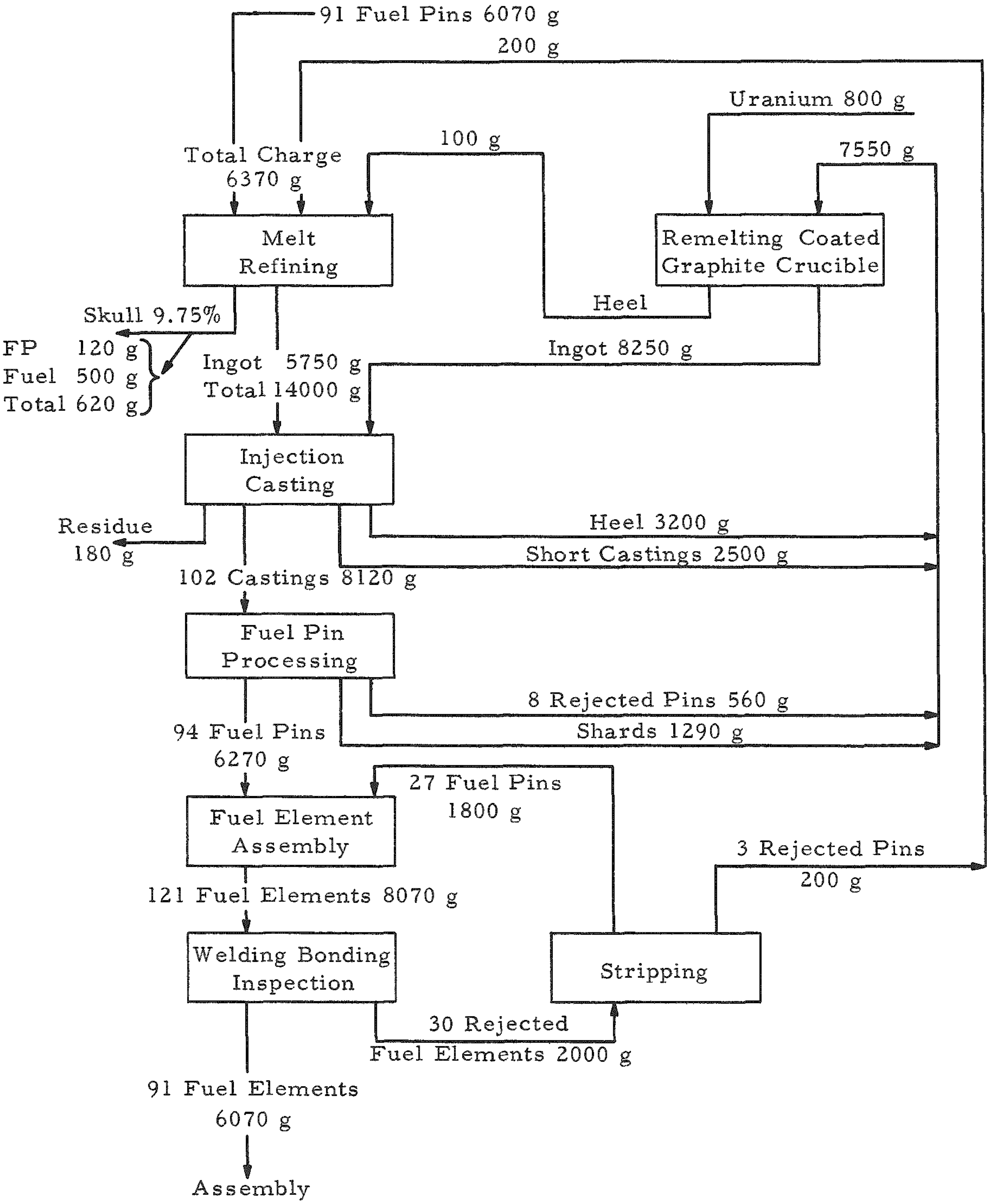


Figure 49

ALTERNATIVE FUEL MATERIAL FLOW DIAGRAM FOR EBR-II FUEL CYCLE - VERSION 3 (No Scrap Remelt - Makeup Fuel and Scrap Added to Melt Refining Crucible)

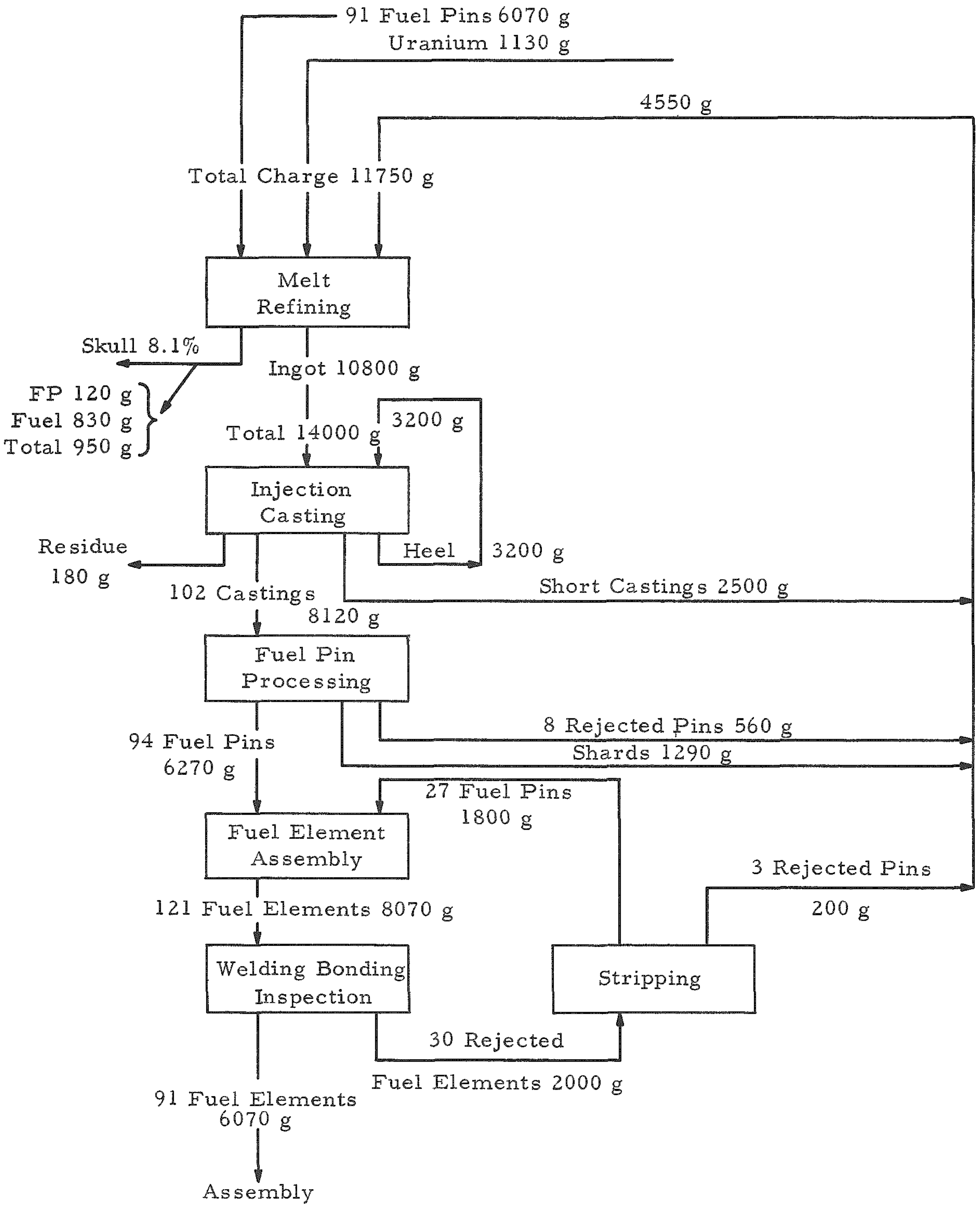


Figure 50

ALTERNATIVE FUEL MATERIAL FLOW DIAGRAM FOR EBR-II FUEL CYCLE - VERSION 4 (Same as Version 3, Figure 49 Except Reduced Charge)

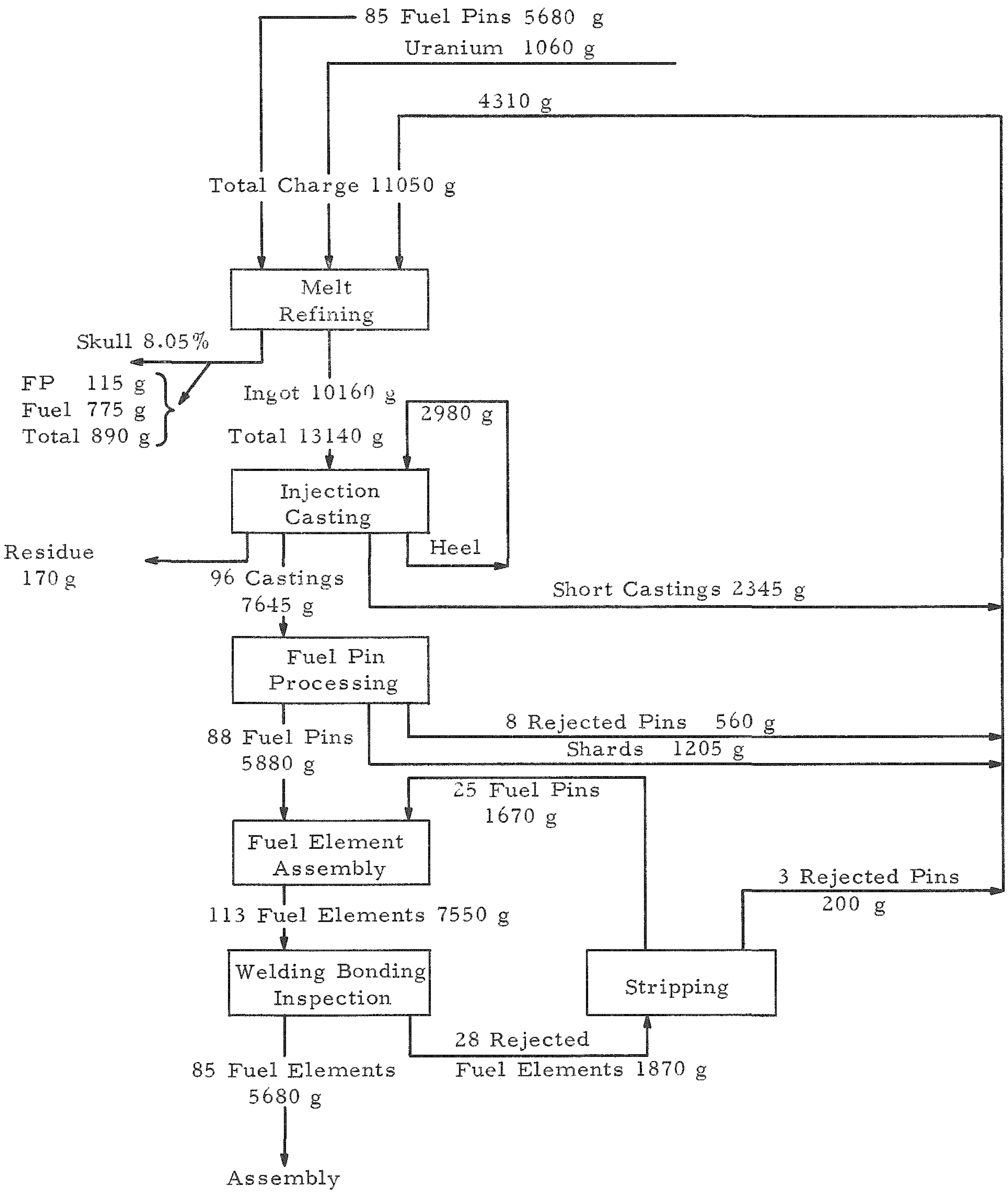
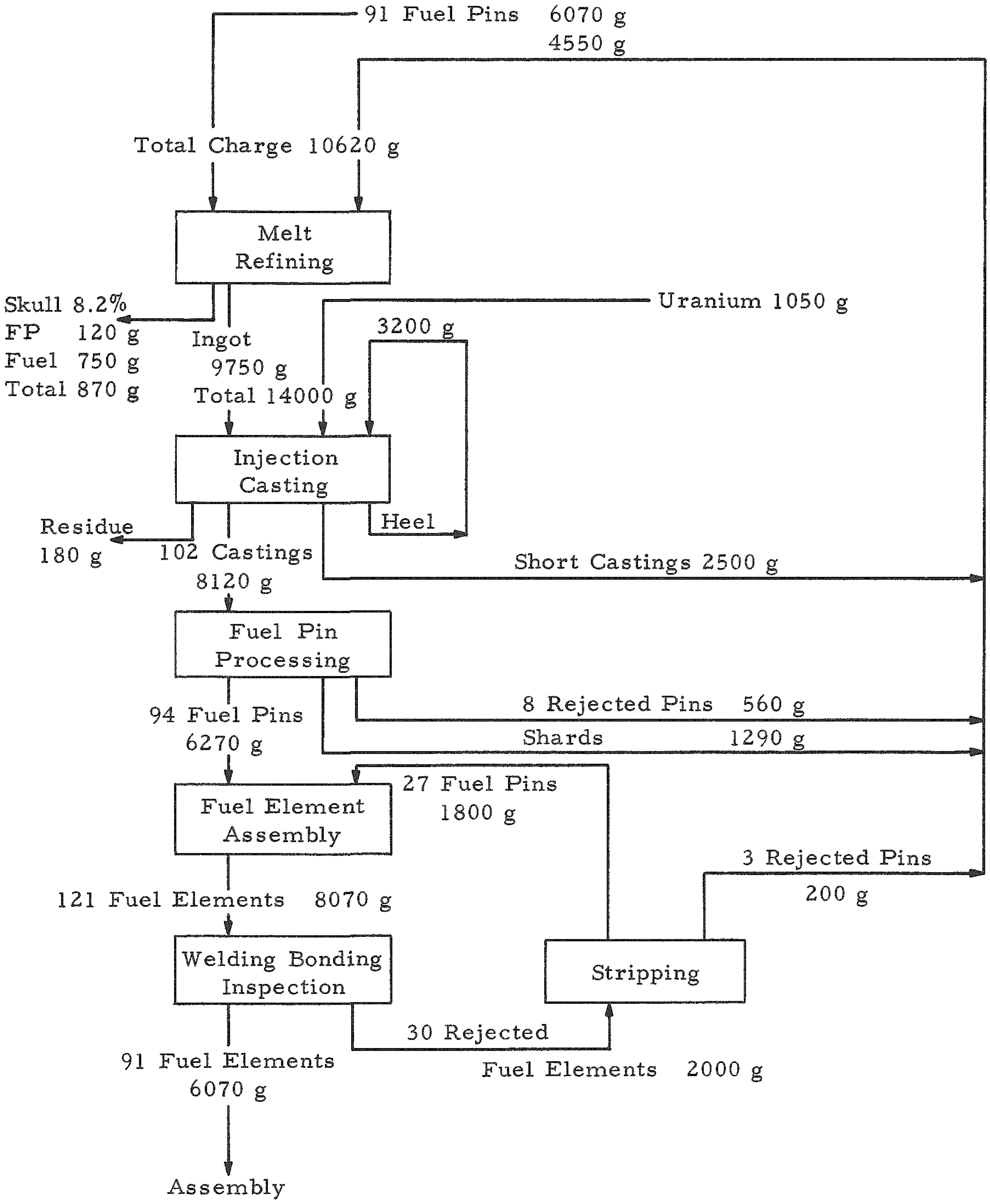


Figure 51

ALTERNATIVE FUEL MATERIAL FLOW DIAGRAM FOR EBR-II FUEL CYCLE - VERSION 5 (Same as Version 3, Figure 49 Except that Makeup Fuel Is Added to Injection Casting)



practice. An inventory of fuel elements would permit lead time in analyses. For instance, as soon as a subassembly was dismantled, one fuel element could be immediately moved into the argon cell for decanning and sampling.

Figure 47 shows a possible flow diagram utilizing scrap remelting; its basis is 91 fuel elements charged and 80 fuel elements produced. In this case, only enriched uranium (to compensate for burnup) would be added, in either the melt refining step or the scrap-remelting step. Eleven new fuel elements prepared outside the processing cells would be added to compensate for the losses in the skull and injection-casting residue.

Figure 48 shows a possible flow diagram based on a 91-fuel element charge and a 91-fuel element output with scrap remelting. Uranium is added at the scrap remelting step to compensate for losses and burnup.

Figure 49 shows a possible flow diagram based on 91 fuel elements charged and 91 produced. In this version, scrap and uranium are added at the melt refining step, and the injection-casting heel is recycled to the injection-casting operation. Figure 50 is the same as Figure 49 except that it is based on 85 elements instead of 91. Figure 51 is the same as Figure 49 except that it is based on adding uranium at the injection-casting step instead of to the melt refining crucible.

Various other combinations of fuel cycles are possible. For example, the injection-casting residue insofar as it is removable from the crucible could be added to skull being stored for later reclamation. Unexpected fuel losses could occur. Some of these are fuel losses in pin-cladding scrap, fuel losses because of adherence to used crucibles, fuel losses due to spills, and fuel losses in mold scrap.

2.4.2 Subassembly Transfer and Storage

Normally, subassemblies will be transferred to and from the reactor as they are needed for disassembly and as they are reassembled. The steps in transferring subassemblies between the reactor and air cell are described under "Fuel Processing Operations and Equipment," Section 2.1, p. 47.

Although there are some air-cooled storage pits in the passageway, the only shielding facilities where active subassemblies can be removed from the coffin are at the refueling machine and at the Air Cell port. If subassemblies were stored in these pits, additional air cooling would probably be required. Short-term or overnight storage of a subassembly in the coffin is possible.

Subassemblies might be stored on a limited basis in the air-cooled and water-cooled pits in the air cell, although additional air cooling would probably be required for spent or refabricated subassemblies.

2.4.3 Fuel-element Transfer and Storage

The fuel elements are normally stored only temporarily. The elements from disassembled subassemblies will be decanned and the pins melt refined as soon as conditions permit. Likewise, refabricated fuel elements will be reassembled into subassemblies as soon as feasible. However, from a cooling viewpoint, it appears that fuel elements in magazines will be easier to store than either subassemblies or unclad fuel, such as pins or ingots. In addition, the fuel in a fuel element is protected from nitridation by the cell atmosphere, whereas unclad fuel is not. It thus appears that the small inventories necessary for the smooth operation of the process should be, when possible, in the form of fuel elements, in preference to unclad fuel or subassemblies.

It is anticipated that at times fuel elements equivalent to about two subassemblies may be awaiting decanning. Fuel elements requiring special handling or disposal, such as badly bowed or damaged elements, may also accumulate. Fuel elements equivalent to two subassemblies may be in the various stages of welding, bonding, testing, and inspection, and fuel elements equivalent to two subassemblies may be awaiting reassembly into subassemblies.

The fuel elements awaiting decanning or disposal are normally stored and transferred in horizontal magazines which hold up to 25 fuel elements each. Four of these magazines are required to hold the 91 elements from one core subassembly. Each magazine is normally attached to a transfer cart except during unloading of the elements at the decanner. About 15 magazines and carts are to be used for storing and moving fuel elements which are awaiting decanning or disposal. In general, the magazines and carts will be kept on tracks at the disassembly station, in the small locks, at the decanning station, or on special storage tracks when not being transferred.

The carts which hold the horizontal magazines are 24 in. long, $12\frac{1}{4}$ in. wide, and $6\frac{1}{2}$ in. high. They are too long to be lowered in a horizontal position into the small locks as are the standard carts. These magazine carts are carried in a vertical position to and from the locks by holding one of the carrying handles (at either end of the cart) with a manipulator. In putting the cart into the lock, it is lowered until the "lower end" wheels touch the track. These wheels then roll on the track into the lock. The cart is lowered further and attains a horizontal position when the "upper end" wheels touch the track.

During the latter stages of reprocessing, when the pins are in the cans, the reprocessed fuel elements are stored and transferred while in "A," "B," or loading magazines. Rejected elements which are scheduled to be decanned are placed in horizontal magazines.

Satisfactory fuel pins are placed in cans and then in the "A" magazine at the magazine loader in the argon cell. The magazine is moved to the pin-settling station. The magazine is then transferred to the welder, where the end caps are welded on. After welding, the container is placed in a standard transfer cart, which is transferred through the small lock to the air cell. In the air cell, the magazine is transferred to the leak-detecting station. Elements passing the test are placed in a "B" magazine. Elements failing the test but suitable for rewelding are replaced in an "A" magazine for return to the argon cell and rewelding. Elements which require decanning are placed in a horizontal magazine for transfer to the decanner.

The "B" magazine is transferred to the bonding and bond-test position and from here to the inspection station. Elements which pass the bond test and inspection are cleaned and placed in loading magazines. Elements failing the test but which are suitable for rebonding are replaced in a "B" magazine. Elements failing because of unsuitable sodium level, persistent failure to bond, visible defects, or other reasons which require decanning are placed in a horizontal magazine for transfer to the decanner. The final loading magazine is transferred to the assembly station for reassembly of the elements into subassemblies.

2.4.4 Fuel Pins and Ingot Transfer and Storage

The decanned fuel pins are not ordinarily stored but are charged to the melt refining crucible immediately after decanning because of self-heating and nitridation problems. It is estimated that $1\frac{1}{2}$ hr are required to decan 91 fuel pins and charge them into the melt refining furnace. At the decanning station, the chopped pins are fed directly to the crucible-charging pan. As soon as the batch of pins is decanned, they are moved to a weighing station and are then charged into the melt refining crucible. The decanning operation is not started until the melt refining furnace is ready to receive a charge.

Normally, the storage of ingots is kept to a minimum because of self-heating and nitridation problems. It may be possible to minimize exposure of the ingots from the melt refining and/or remelting steps by moving them directly to the injection-casting crucible. If ingots are to be stored for more than several hours, it may be necessary to move them to a special cooling location or to a special container with bell jar cover where an atmosphere very low in nitrogen can be maintained.

2.4.5 Recycle Scrap Transfer and Storage

The recycle scrap consists essentially of heels, short castings, rejected pins, and shards from the injection-casting and pin-processing operations. Other miscellaneous items may be included. The short castings, rejected pins, shards, and any other small pieces of fuel may be added to either a melt refining charge or to an injection-casting charge. During Core I production, in which pins were cast from unirradiated uranium-fissium alloy, the hells were recharged to the injection-casting operation.

The short castings, rejected pins, and shards are recovered over a period of a few hours during pin processing. They are placed in a charging container for movement to the remelting crucible. An injection-casting heel is broken into several pieces in the heel breaker for charging into the remelting or injection-casting crucible. Since a furnace may not be available for charging the scrap and heels when they become available, it may be desirable, because of nitridation problems, to have a specially cooled storage location or a special container with a bell jar cover where an atmosphere very low in nitrogen can be maintained.

2.4.6 Skull or Skull Oxide Transfer and Storage

The skulls from the melt refining operation may be handled for storage in one of two ways:

1) the skulls may be stored in the zirconia crucibles in which they are formed, or

2) the skulls may be oxidized to skull oxide and poured from the crucible into a suitable storage container.

When the crucible is removed from the melt refining furnace, it should be placed in a stainless steel crucible holder or can whether or not the skull is to be oxidized. This action should maintain the integrity of the crucible, which is subject to breaking and falling apart. Spring tongs are used when the crucible is moved by the manipulator.

If the skull is not to be oxidized, the crucible is transferred in a covered can to a rack in one of the storage pits. If the skull is to be oxidized, the crucible is transferred to an oxidizing chamber (see Figure 25, p. 75) in which the skull is oxidized by an argon-oxygen mixture. After oxidation, the skull oxide is dumped into a covered storage pan which will be transferred to a rack in one of the storage pits.

There are seventeen water-cooled storage pits in the argon cell. The pits are 12 in. in diameter and 122 in. long inside. They extend $2\frac{1}{2}$ in. above the cell floor and have 65-in.-long water jackets. The water jackets do not enclose the top $53\frac{1}{2}$ in. or the bottom $3\frac{1}{2}$ in. of the pits.

Racks for each storage pit might be designed that will hold about 8 skulls in the crucibles enclosed in cans or about 12 skulls as oxide in pans. In placing the skulls or oxide in the racks, it would be desirable to put the most recently processed material in the water-jacketed section of the pit and to space the items. As the fission product elements in the skulls or oxide decayed and the heating became less, more items could be added, and some of the oldest could be moved to the unjacketed sections.

It appears that skull oxide can be stored more favorably in large-diameter short cans than in small-diameter long cans from both heat transfer and filling viewpoints. For example, cans 6 in. in diameter and 3 in. high would easily hold 800 g of skull oxide (less than one inch of oxide in the bottom).

2.5 Fuel and Waste Sampling and Analyses

2.5.1 Sampling Points

The proposed analytical work and procedures are discussed in reports by Vogel(77) and McCown.(78) In these reports the sample points are numbered as follows (the fuel sample points 1-11 are shown in Figure 52):

Fuel Processing

1. Stainless steel pin-cladding scrap
2. Decanned fuel pins
3. Melt refined ingot
3a. Remelted ingot (remelted scrap)*
4. Melt refining skull
4a. Oxidized skull*
5. Recovered uranium from skulls
6. Enriched uranium from AEC
7. Off-gases from melt refining holdup tank
8. Slag from injection-casting process
8a. Vycor from injection-casting molds*
9. Waste from skull reprocessing
10. Injection-cast pins
11. Used crucibles from melt refining and injection casting

Blanket Processing

12. Decanned blanket pins
13. Uranium from blanket processing
14. Magnesium with plutonium from blanket processing
15. Distilled magnesium from blanket processing
16. Plutonium from distillation

Argon and Sodium (Reactor and Fuel Cycle Facility Operation)

17. Argon from disassembly cell at the reactor (now eliminated)

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*These sample points were not listed in the reports cited above.

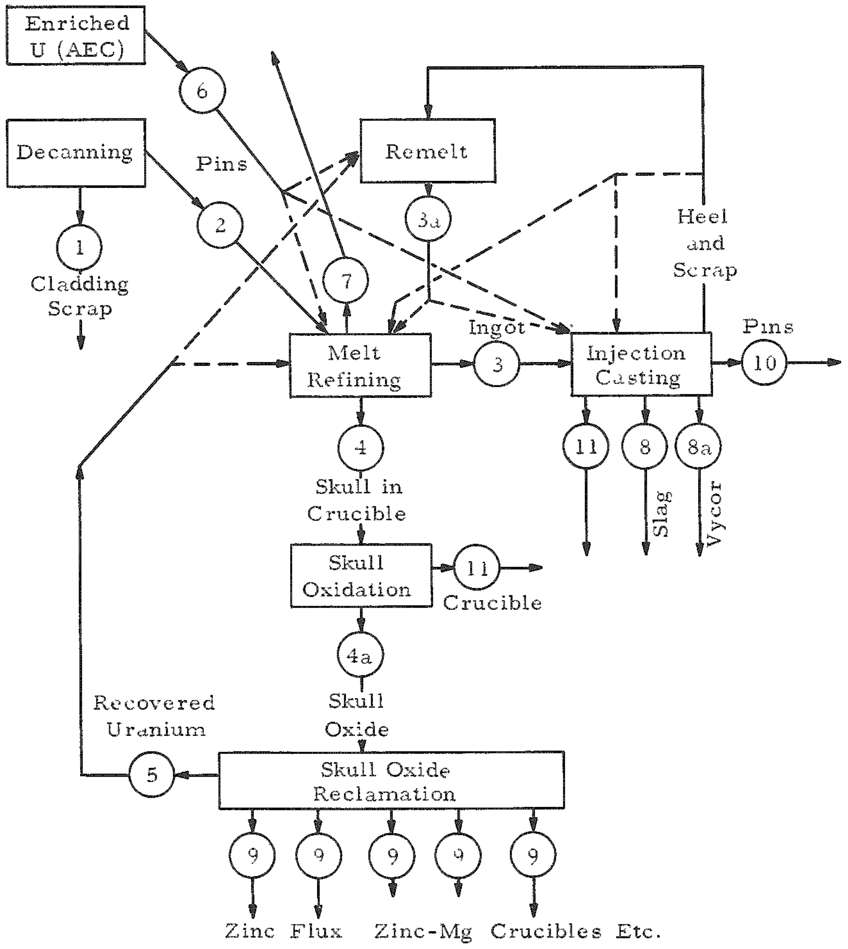
18. Argon from reactor
19. Argon from cell to stack (now eliminated)
20. Argon to disassembly cell at the reactor (now eliminated)
21. Argon from reactor to stack
22. Argon to stack
23. Argon from cell to purifier
24. Argon from purifier to cell
25. Sodium (oxide) from sodium oxide-removal equipment in reactor building
26. Sodium from reactor to sodium oxide-removal equipment
27. Sodium from reactor primary tank
28. Sodium from secondary system
29. Sodium (oxide) secondary cleanup system

Waste Streams

30. Alkaline aqueous waste streams

Figure 52

MAJOR FUEL SAMPLE POINTS IN THE EBR-II FUEL CYCLE



Sample points 1 to 11 involve fuel processing; 12 to 16 involve blanket processing; 18, 21, 25, 26, 27, 28, and 29 involve reactor operation; 17, 19, and 20 have been discontinued; and 22, 23, 24, and 30 involve the Fuel Cycle Facility operation. In this discussion, means of obtaining samples and possible analyses required for 1 to 12, 22 to 24, and 30, which involve the Fuel Processing Facility, will be discussed.

2.5.2 Pin-cladding Scrap, Sample Point 1

The purpose of sampling the cladding scrap is to check it occasionally for possible losses of uranium fuel. Because any uranium fuel in the scrap might be distributed in a nonuniform manner, it is desirable that the scrap from a number of fuel elements be treated to obtain a representative sample. The cladding scrap consists of about 820 g of decanning ribbon, 45 g of bonding sodium, and 865 g of spacer wire and ends per subassembly of 91 fuel elements. For 2 percent burnup and 15-day cooling, it is estimated that the activity of the stainless steel scrap would be about 3 c/g. The activity due to 15-hr sodium-24 would be quite low (less than 1 mc/g of sodium). The activity of any fuel accompanying the scrap would be about 45 c/g. To remain below the 10-c-per-sample limitation mentioned in the Analytical Flowsheet reports,(77,78) the size of samples which could be taken to the analytical caves in the Laboratory and Services Building would be small.

Sampling of the scrap will be difficult. Several proposed methods for obtaining a sample are as follows, in order of apparent feasibility:

1) Melt the stainless steel scrap in the melt refining furnace, pour it into a mold, and take samples from the poured ingot, possibly by breaking off projections cast on the bottoms of the ingots.

2) Submit the ribbon scrap of about 9 g from one fuel element for analysis.

3) Dissolve any fuel from the scrap in acid in the air cell, and take liquid samples; however, no provisions for handling aqueous solutions in the air cell exist.

4) Dissolve the scrap in a molten metal (such as zinc) in the argon cell, and take liquid samples. It is not known what temperature would be needed to keep uranium in solution in the presence of stainless steel.

Only occasional sampling of the scrap for uranium analysis is contemplated, and only the ribbon scrap need be sampled, since the spacer wire and ends should not contain lost fuel.

2.5.3 Decanned Fuel Pins, Sample Point 2

This sample will be used to determine the characteristics of reactor-discharged fuel. Analyses to determine uranium-235, rare earths, and burnup may be made. The sample will be obtained by breaking off about $\frac{1}{4}$ in. of a decanned fuel pin. The sample will weigh about 1.2 g and will have an activity of approximately 55 c for 2 percent burnup and 15-day cooling. The sample will be placed in a $\frac{3}{4}$-in.-diameter by 2-in.-long, aluminum sample can for transfer to the analytical caves.

Any predetermined section of a pin from a predetermined fuel element from a subassembly may be selected for the sample. Each subassembly may be placed in the reactor core in either of two angular orientations, which are diametrically opposed; however, the reactor operators can determine this orientation with respect to the orientation of a subassembly in the coffin. The position in the reactor of the center fuel element of a subassembly is independent of angular orientation; hence this element may be most useful for analyses.

Some reassembled subassemblies may be provided with one or more new fissium fuel elements or with special fuel elements. Special samples may be required from fuel elements in such subassemblies.

2.5.4 Melt Refined Ingot, Sample Point 3

This sample will be used to obtain analyses of the melt refined fuel. Analyses to determine uranium-235, uranium-238, uranium-236, uranium-237, total uranium, plutonium-239, neptunium-237, zirconium, molybdenum, technetium, ruthenium, rhodium, palladium, rare earths, barium, lanthanum, and other fission products, and iron, silicon, and thorium may be needed.

The samples will be obtained by breaking off two $\frac{1}{4}$-in.-long by 0.145-in.-diameter projections from the bottom of each melt refining ingot. The samples will weigh about $1\frac{1}{4}$ g each and will have an activity of about 15 c each for 2 percent burnup and 15-day-cooled fuel. Each sample will be placed in a small aluminum sample can for transfer to the analytical caves.

2.5.5 Remelted Ingot, Sample Point 3a

This sample will be used to obtain analyses of remelted ingots from injection-casting scrap, when desired. The process of obtaining samples will be the same as in the case of melt refined ingots.

2.5.6 Melt Refining Skull, Sample Point 4

This sample will be used in making an occasional check of the slag from the melt refining operation before it is oxidized. Analyses for total uranium, plutonium-239, and various fission product elements may be required.

A representative sample will be difficult to obtain. The skull is attached to the crucible; however, pieces can usually be broken off. A mortar and pestle to be operated within the argon cell or the air cell would probably be necessary to obtain a representative sample of the pieces which are broken off. These pieces, however, may not be representative of the total skull. The activity of the skull material from 2-a/o-burnup, 15-day-cooled fuel may be as high as 275 c/g; however, additional cooling time could be allowed before the samples are sent to the analytical caves.

2.5.7 Oxidized Skull, Sample Point 4a

The purpose of this sample is to make possible various analyses of the oxidized skull material from a melt refining operation. Analyses would be similar to those listed under melt refining skulls. The sample can be obtained from the granular, oxidized skull, as it is poured into the storage pans. Since tests with nonirradiated fissium-uranium alloys show some differences in analyses between coarse and fine materials, samples of both coarse and fine materials may be desired. A tight sample container which fits inside a sample can may be desirable since a fine active material will be involved. As pointed out above, the activity of the skull oxide material from 2-a/o-burnup, 15-day-cooled fuel may be as high as 275 c/g, but additional cooling time could be allowed before the samples are sent to the analytical caves.

2.5.8 Recovered Uranium from Skulls, Sample Point 5

The process requiring this sample is under development, and analyses have not been specified. They would, however, include analyses for total uranium content, molybdenum, rhodium, ruthenium, palladium, magnesium, and zinc.

2.5.9 Enriched Uranium from AEC, Sample Point 6

The purpose of this sample is to make check analyses of the enriched uranium which will be added to the process. Analyses for uranium-235, uranium-236, and uranium-234 may be required. Sampling would be straightforward.

2.5.10 Off-gases from Holdup Tank, Sample Point 7

The purpose of this sample point is to monitor the activity of krypton, xenon, and possibly iodine as it is being released to the stack. A continuous-monitoring instrument is planned.

2.5.11 Slag from Injection-casting Process, Sample Point 8

This sample is used to make occasional analyses of the fine residues from the injection-pin-casting and pin-preparation operations. In work with unirradiated fuel, this residue has amounted to about 2 g per fuel element processed. Analyses for uranium and thorium may be required. Thorium oxide is used as a coating for the crucibles and molds. Samples could probably be obtained by quartering the material. Activities up to about 15 c/g could be expected.

2.5.12 Vycor from Injection-casting Molds, Sample Point 8a

Occasionally, the Vycor scrap from the injection-casting molds will be sampled to check for uranium losses. Obtaining a suitable sample will be difficult. Quantities of the broken Vycor could be pulverized by a mortar and pestle and quartered for a sample. The work could be done in the air cell.

2.5.13 Waste from Skull Processing, Sample Point 9

The process requiring this sample is under development; however, sampling of waste halide flux, zinc, zinc-magnesium mixtures, and discarded crucibles will probably be required.

2.5.14 Injection-cast Pins, Sample Point 10

This sample will be used for analyses of fuel being returned to the reactor. Determination of enrichment will be important. Analyses to determine uranium-235, total fissionable material, and other components may be necessary.

The sample will be obtained by taking part of an injection-cast pin and shearing off $\frac{1}{4}$ in. for the sample. The sample will weigh about $1\frac{1}{4}$ g and have an activity of about 15 c for 2-a/o burnup and 15-day cooling. It will be enclosed in a $\frac{3}{4}$-in.-diameter by 2-in.-long, aluminum sample can for transfer to the analytical caves.

2.5.15 Used Crucibles from Melt Refining and Injection Casting, Sample Point 11

The purpose of this sample is to check occasionally the melt refining and injection-casting crucibles for uranium fuel losses. Total uranium analyses would be made.

Obtaining a suitable sample will be difficult. One of the following sampling methods might be used:

1) Scrap the material from the interior of the crucible and quarter it for a sample.
2) Leach the crucible with acid and take an aqueous sample; however, no provisions for handling aqueous solutions exist in the air cell.
3) Crush the crucible, and quarter it for a sample.

The sampling work could be done in the air cell.

2.5.16 Decanned Blanket Pins, Sample Point 12

This sample is used in determining the composition of blanket material which is discharged from the reactor. Analyses to determine plutonium and burnup will be required. No provisions for taking samples from the blanket elements have been planned.

The upper and lower blanket elements from the core subassemblies will probably be the first blanket material to be obtained from the reactor. At 2 percent burnup of the fuel and 15-day cooling, the end of the pin next to the core would contain about 0.2 percent plutonium and have an activity of about 2 c/g. This pin would have a total plutonium content of about 0.1 g and a total activity of about 100 c. The plutonium content and activity of the second pin from the core would be relatively low.

2.5.17 Argon to Stack, Sample Point 22

The purpose of this sample is to check the argon for activity being released to the stack. A continuous-monitoring system is planned.

2.5.18 Argon from Cell to Purifier, Sample Point 23

This sample will be representative of the argon in the argon cell, its purpose being to check occasionally the cell atmosphere for oxygen, nitrogen, and water vapor. Analyses for activity and other items may also be made.

2.5.19 Argon from Purifier to Cell, Sample Point 24

The purpose of this sample is to check the argon from the oxygen- and water vapor-removal equipment for oxygen, nitrogen, and water vapor content. Continuously analyzing monitors for oxygen, nitrogen, and water vapor are planned. The oxygen analyzer is based on an electrolytic cell output, the nitrogen analyzer on thermal conductivity, and the water vapor analyzer on the electric current required to decompose the water.

2.5.20 Alkaline Aqueous Water Stream, Sample Point 30

Radiation monitoring of the wash water during subassembly cleaning in the coffin should indicate any substantial loss of integrity in fuel-element cladding. Serious conditions would possibly have been detected by the reactor operators. Sampling of the wash water and retention-tank contents will depend upon conditions which develop during actual operation. For criticality control, sampling of retention tank liquids for fissionable material content will be necessary.

2.6 Feed Materials Preparation, Handling, and Storage

Provisions have been made in the Fuel Cycle Facility for handling and preparing various consumable materials, including crucibles, molds, and fume traps. The auxiliary cell equipment required for these functions includes the following items:

1) a spray booth in the mold-preparation room for coating molds and crucibles with thoria slurry;
2) an induction-heated degassing furnace, with vacuum pump in the degassing room for degassing molds, crucibles, and fume traps;
3) a degreasing vat in the dry-box room for degreasing subassembly parts and fuel-element cans;
4) a sodium glove box in the dry-box room for adding sodium to the fuel-element cans.

2.6.1 Melt Refining Crucibles

The zirconia crucibles will be received from the manufacturer packed in boxes. After inspection, they may be replaced in the shipping boxes for storage. Before use, the crucibles will be degassed and inspected. Tests are being conducted to determine the rate of re-adsorption of gases and vapors from the atmosphere by degassed crucibles. The results of these tests should indicate whether or not airtight bagging is necessary after degassing. Bags could be removed in the air cell by master-slave manipulators, but removal in the argon cell would be difficult.

The crucibles, which are $6\frac{3}{8}$ in. in diameter and $9\frac{1}{2}$ in. high, can be moved into the air cell through the wall port and into the argon cell through the small locks.

2.6.2 Melt Refining Fume Traps

The Fiberfrax fume traps received from the manufacturer in boxes may be stored in the shipping boxes after receipt and inspection. Before use, the traps will be degassed at about 700°C. Preliminary indications are that bagging will not be necessary for these units.

The fume traps, which are 10 in. in diameter and $5\frac{1}{2}$ in. high, may be too large to go into the air cell through the small port. If field tests indicate that this is the case, use will be made of the large port. The traps can be moved into the argon cell through the small locks, although a special carriage may be necessary. It would be possible to transfer the traps into the argon cell inside the cans in which they would later be placed for disposal.

2.6.3 Melt Refining Graphite Ingot Molds

After receipt and inspection of the graphite ingot molds, they may be stored in the shipping boxes. No degassing of the molds is planned before their use, because degassing tends to open pores or voids which cause ingot sticking.

The graphite molds ($6\frac{1}{8}$ in. high and $4\frac{15}{16}$ in. square) can be moved into the air cell through the small port. They are assembled in the air cell into the steel mold holders, which are $5\frac{1}{2}$ in. by $5\frac{3}{4}$ in. by 7 in. high. Transfer of these assembled units to the argon cell can be made through the small locks. Used and damaged molds are removed from their holders in the air cell.

2.6.4 Injection-casting Graphite Crucibles

The graphite crucibles may be handled in the same manner as other crucibles and molds, i.e., inspected and then stored in their shipping boxes. Before use, the crucibles will be coated, on the outside as well as the inside, with a mixture of 95 percent thoria and 5 percent zirconia. The outside coating reduces heat loss by radiation; the inside coating prevents reaction of molten uranium with graphite. About 35 g of thoria are used per crucible.

During the coating process, the crucibles are first brushed or wiped to remove loose material. They are then sprayed with a water slurry of the thoria and zirconia having a particle size of minus 325 mesh. The crucibles are then oven-dried.

The crucibles ($6\frac{3}{4}$ in. in diameter and $5\frac{3}{4}$ in. high) can be moved into the air cell through the small port and into the argon cell through the small locks.

2.6.5 Injection-casting Pallets

The injection-casting pallets for holding the pin molds are made of steel. No cleaning or preparation before their use is anticipated.

The pallets are 6 in. in diameter and $20\frac{1}{8}$ in. long. The molds in a pallet extend about $4\frac{1}{2}$ in. below the bottom of the pallet, but they are free to move upward in the perforated plates and could be broken by rough handling Pallet loading will be done in the argon cell by use of the sealed Model "A" manipulator.

2.6.6 Injection-casting Pin Molds

Before use, the Vycor molds are coated inside with thoria or a 98 percent thoria-2 percent kaolin mix. About $\frac{1}{2}$ g of thoria is used per mold. The molds are coated in the mold booth by swabbing with a slurry of minus 325-mesh thoria (and kaolin, if used) in methyl alcohol. A pipe cleaner attached to a rod has been used as a swab. The molds are air-dried for several hours, with the open end down, and then oven dried at about 600°C for 4 to 6 hr with slow heating and cooling. The molds are then kept in a drying oven at about 150°C until they are sent into the argon cell for use.

The Vycor molds, which are about $18\frac{3}{8}$ in. long and $\frac{15}{64}$ in. in outside diameter, can be readily sent into the air cell through the small port in a suitable container; however, their transfer into the argon cell through the small air locks will require care.

2.6.7 Injection-casting Heel Filters or Remelt Crucibles

If graphite filter crucibles are used to remelt and filter the heel material from injection casting, or if graphite remelt crucibles are used in the melt refining furnace to remelt heels and scrap, they would probably be coated with thoria and handled similarly to injection casting crucibles.

2.6.8 Cladding Cans and Restrainers

Before their use, the restrainer caps are cleaned with a chlorinated hydrocarbon, such as trichloroethylene, and the cladding cans are charged with sodium. The $2\frac{3}{4}$-in.-long restrainers are cleaned, dried in the degreaser, placed in a suitable container from which they can be charged into the welder hopper, and then sent through the small air cell port and through a small lock into the argon cell.

It is anticipated that the cladding cans will not require cleaning. The interior diameters of the cans are measured with an air gauge, and they are sorted into ten groups based on inside diameter and, hence, on the amount of the sodium required.

The sodium is placed in the cans in the inert-atmosphere glove box. Sodium is extruded in the glove box into a rod about $\frac{1}{8}$ in. in diameter, cut into pieces, and weighed into ten groupings for the respective can groups. The amount of sodium required is about 0.8 g per can. The required weight of sodium is placed in each can, and the cans are placed in a meltdown furnace to melt and consolidate the sodium at the bottoms of the cans. After the cans are cooled, they are corked and kept in the inert atmosphere until they are sent into the cells.

The cans, which are about 18 in. long, are sent into the argon cell through the small port and small air locks in a suitable container. The corks may be removed from the cans in the argon cell by the master-slave manipulators. If a cork is dislodged during evacuation of a small air lock containing corked cans, this should cause no difficulty. If necessary, corks may be removed in the air cell just before the cans are sent through the air locks.

2.6.9 Hex Tubing for Subassemblies, Top Fixture with Upper Blanket Elements, and Bottom Adapter with Bottom Blanket Elements

It is anticipated that the hex tubing and the top fixture with upper blanket elements will be assembled by the manufacturer before shipment. After receipt and inspection, these items will probably be stored in a manner suitable for accountability purposes because of the depleted uranium contained in the blanket elements.

Because of their length, it will probably be necessary to send these items into the air cell through the large port.

2.6.10 Control and Safety Subassemblies

Since the control and safety subassemblies are smaller and have only 61 fuel elements and no blanket elements, they might be replaced as new units instead of reassembling this smaller size of subassembly in the air cell. These subassemblies would be processed, but the container fuel would be used in making regular core subassemblies. All subassemblies (control, safety, and core) would need to be stored in a manner suitable for accountability purposes.

During transfer of new subassemblies to the reactor, they could be loaded into the coffin outside of the air cell, provided the coffin were not severely contaminated; if the coffin was contaminated, the subassemblies would be sent into the air cell and there loaded into the transfer coffin through the large floor port.

2.6.11 Fuel Elements and Fuel Material

Some fuel may be added to the cycle as fuel elements. In this case, the fuel elements could be transferred into the air cell through the small port.

Uranium-235 and perhaps natural uranium and fuel alloy or noble fission elements will be added to the cycle to compensate for burnup and processing losses. These materials could be transported in melt refining crucibles or in other suitable containers.

It is necessary that fuel elements and fuel material be stored in a manner suitable for accountability purposes.

2.6.12 Miscellaneous

Miscellaneous items such as melt refining furnace filters, thermocouples, susceptors, and equipment repair parts, will be sent into the cells from time to time.

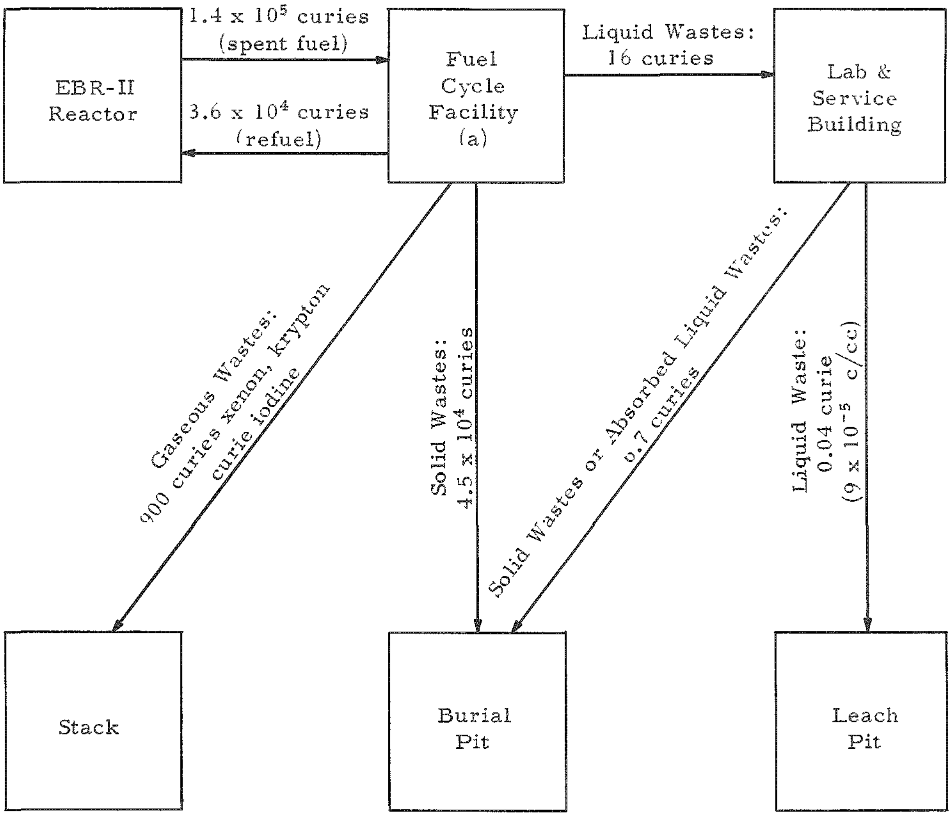
2.7 Waste Materials: Handling and Storage

Liquid, solid, and gaseous wastes must be handled in connection with the operation of the Fuel Cycle Facility. Numerous safeguards have been included in the design to minimize the possibility of the inadvertent release of activity to the environments. These include containment of processing facilities in closed, inert-atmosphere cells, the use of a large holdup tank for off-gases in order to permit dilution and controlled release of gaseous activity to the atmosphere, and the employment of filter systems to remove particulate matter before discharge of gas from the 200-ft stack.

The major waste streams for the EBR-II fuel cycle are shown in Figure 53. It may be noted that about one-third of the radioactivity in the

Figure 53

OVERALL DAILY WASTE FLOW FOR EBR-II FUEL CYCLE(79)



(a) The difference between daily input and output curies is due to decay in the Fuel Cycle Facility.

spent fuel is recycled in the refabricated fuel. Most of the radioactivity removed in processing is discarded in solid wastes (used crucibles, fume traps, and waste zinc or zinc alloys from the Skull Reclamation Process). The fission product rare gases, xenon and krypton, are the principal radioactive contaminants of the gaseous wastes. Iodine is also a potential contaminant in gaseous wastes, but considerable effort is made to reduce to a very low level the iodine activity in the gas discharged from the suspect stack. The liquid wastes are generated principally in the removal of sodium from fuel subassemblies and in analytical operations. In normal operation, little radioactivity will be present in liquid wastes, but in the event of a ruptured fuel element, the activity level in liquid wastes could become considerably higher than that shown in Figure 53. Solid wastes consist principally of cladding scrap, used crucibles, molds, crushed Vycor tube molds, fume traps, and miscellaneous items of cell equipment. The handling of gaseous, solid, and liquid wastes are discussed individually in the following sections.

2.7.1 Gaseous Waste Disposal

The following are the principal sources of gaseous waste in the Fuel Cycle Facility:

1) gases evolved or released during the melt refining operation;
2) off-gases from the oxidation of the skull or residue material remaining in a melt refining furnace;
3) gas withdrawn from the argon cell in order to maintain the cell gas pressure negative with respect to the air pressure in the operating areas. The necessity for such adjustments in the argon cell pressure results from atmospheric pressure fluctuations. The removal and addition of argon to the argon cell to maintain the proper pressure is called "cell breathing";
4) gases evacuated from transfer locks and from process equipment other than the melt refining and skull oxidation furnace, for example, from the injection-casting furnace;
5) air-cell ventilating air (5000 cfm);
6) air exhausted from the sodium-disposal box in the decontamination room and through the 4-in. sodium wash water drain in the interbuilding passageway;
7) exhaust ventilation air from offices and laboratories.

The major source of high-activity gaseous waste is the melt refining and skull oxidation operations. The fission product rare gases, xenon and krypton, are released in the melt refining operation. Iodine is a potential radioactive contaminant of the exhaust gases from both the melt refining and skull oxidation operations. These gases receive special treatment before disposal to the atmosphere in order that the concentration of fission product gases in the atmosphere may be maintained within established safe limits.

The other sources of gas are not expected to contain significant amounts of radioactivity, and any radioactivity present is expected to be particulate in nature. These gases are, therefore, simply combined with the building exhaust air, filtered, and disposed of through the 200-ft suspect-gas stack. Slightly over 60,000 cfm of the gas discharged through the stack consists of exhaust gases from the Reactor Building, the Fuel Cycle Facility, and the Laboratory and Service Building. This large quantity of gas will serve to dilute appreciably any activity present in any gas source.

There are two exceptions to this method of gas handling. The first is that the gas involved in sodium-disposal operations is drawn by a compressor through a venturi scrubber to remove any sodium present, and then through an entrainment separator, a heater, and an AEC filter before being discharged by the compressor to the building ventilation system. The second exception is that in an emergency caused by rapid fluctuations of atmospheric pressure or by failure of the cooling system for the argon cell gas, the argon cell gas may be discharged directly to the atmosphere through valves located on the building roof. Normally, it is exhausted directly through filters to the suspect-gas stack if its pressure becomes higher than a negative $\frac{3}{4}$ in. of water pressure with respect to pressure in the operating areas. The emergency release of argon directly to the atmosphere would occur if the argon cell pressure became about one inch of water pressure higher than the operating area pressure.

2.7.1.1 Disposal of High-activity Gases from Melt Refining and Skull Oxidation Operations

The principal source of high-activity gaseous wastes is the melt refining operation. The fission product gases, xenon and krypton, are released during the melt refining step. Iodine also is volatilized in melt refining, but a large fraction of it appears to condense in the melt refining furnace. In Table 14 are shown the rate of generation (c/day) of iodine, xenon, and krypton present in discharged EBR-II fuel (after 2-a/o burnup in 135 days and after 15 days of cooling) and the permissible discharge rates of these fission products from the stack. It may be seen that to remain within permissible concentrations at ground level, reduction factors of 4×10^{12} and 1×10^{10} are required for iodine and xenon, respectively. As pointed out in the footnote in Table 14, the reduction factor required for krypton is about 10^7.

Table 14

FACTORS PERTINENT TO DISCHARGE OF GASEOUS ACTIVITY FROM SUSPECT STACK(81,82)

Basis: Continuous fission of U^{235} at the rate of 62.5 g/day,
for 135 days, followed by 15 days of cooling.

	Iodine		Xenon		Krypton	
Quantity	I^{131}	Total Iodine	Xe^{133}	Total Xenon	Kr^{85}	Krypton
Atoms/100 atoms fissioned	0.073	1.059	0.064	21.775	0.264	2.915
Gram-atoms/day	1.94×10^{-4}	2.82×10^{-3}	1.70×10^{-4}	5.78×10^{-2}	7.01×10^{-4}	7.74×10^{-3}
Grams/day	-	0.36		8.13		0.72
Curies/day	3.14×10^3	-	4.4×10^3	-	35	-
Curies at Equil.	3.65×10^4 [a]	-	3.38×10^4 [a]	-	(1.73×10^5)[a]	-
Maximum Permissible Concentration at Ground Level, Curies/cu meter	9×10^{-9} [b]	-	3×10^{-6} [b]	-	3×10^{-6} [b]	-
Factor of Reduction Necessary	4×10^{12} [a]	-	1.1×10^{10} [a]	-	[a]	-

[a] These values give the maximum amount of isotope that could be present on the basis that decay is the only method of removal. For iodine and xenon, *which have short half-lives, these values are probably close to the total in the system. For krypton, which has a long half-life (9.4 yr), the value has* little significance. For krypton, the required factor of reduction in concentration will be about 10^7.

[b] Federal Register, November 17, 1960; Title 10 - Atomic Energy, Chapter 1, Atomic Energy Commission, Part 20, Standards for Protection Against Radiation.(83) These maximum permissible concentration values are based on 40 hr of human exposure in any period of seven consecutive days.

Considerable reduction in concentrations at ground level is realized by discharge of the gases from a 200-ft-tall stack. Because the radioactive gases will be discharged under favorable meteorological conditions, i.e., adequate wind velocity and absence of inversions, it is proper to take into account the concentration decrease from stack to ground level. For an average wind velocity of 8 miles per hour, the concentration at ground level is reduced by a factor of 1800,(80) thereby reducing the required concentration reduction factors to 2.2×10^9 for iodine and 6.1×10^6 for xenon. The required concentration reduction factors for these elements are achieved in different fashions.

Xenon and Krypton

Dilution with stack gas is the method employed to reduce the xenon concentration in gaseous wastes to an acceptable level. The volume of the holdup tank (500 ft^3) was selected to provide approximately a 30-day holdup of gases discharged from the melt refining furnace. Thus, about $\frac{1}{30}$ th of the volume, or about 17 ft^3, must be discharged each day if the tank is at full capacity. At xenon equilibrium, this volume would contain $\frac{4.4 \times 10^3 \times 17}{500\lambda + 17}$ or 900 c of xenon activity, where λ is the decay constant ($days^{-1}$). The time over which it could be released without exceeding a ground level concentration greater than 1 MPC* (40 hr) is:

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* Maximum permissible concentrations (MPC) are those concentrations to which a person may be continuously exposed for a 40-hr period in any one week. If the exposure time is less, the maximum permissible concentration may be correspondingly increased. Thus, if the exposure is for only one hour in a week, the MPC may be increased by a factor of forty. This factor was not taken into consideration in the calculation of the period of time over which gas could be released from the holdup tank. Therefore, the release time could be reduced considerably, if desired.

$$\frac{900\ \text{c}}{3 \times 10^{-6}\ \text{c/m}^3 \times 1800} \times \frac{1}{1700\ \text{m}^3/\text{min}^*} = 98\ \text{min.}$$

This calculation simply indicates that the actual daily time required for bleeding xenon activity into the stack gas is small (about 2 hr). The time can be adjusted for different wind conditions, for other operational variables (e.g., fuel burnup and decay time), and on the basis of experience.

Since xenon and krypton behave alike, the required reduction factor of 10^7 for krypton will have been exceeded when a concentration reduction factor of 10^{10} is achieved for xenon.

Iodine

A similar calculation in which the total equilibrium amount of iodine in the system is used shows that dilution by stack gases is inadequate for reducing iodine concentration to acceptable levels. Consequently, other iodine-removal procedures must be employed. Fortunately, it has been found that most of the iodine is condensed within the furnace. In high-activity-level melt refining experiments,(36c) it was found that less than 0.005 percent of the iodine activity originally in the charge appeared in the furnace argon atmosphere after melt refining and cooling of the furnace to equilibrium at ambient temperatures.

In a separate experiment,(40) it was found that iodine of the order of 10^{-3} percent in the charge was removed by pumping out the furnace to a final pressure of 8.5 mm. These gases were drawn through a trap consisting of a seven-section bed of charcoal and two AEC filters (one placed on the fore side and one on the aft side of the charcoal bed to remove any particulate-carried iodine). The effluent from this trap contained iodine activity of the order of 10^{-6} percent or less of that originally present in the furnace. No iodine activity could be detected beyond the first section of the charcoal bed. It is noteworthy that a few tenths of a percent of rare gas activity was retained on the charcoal. This could affect the ultimate disposition of the charcoal trap.

Deposition of iodine within the furnace can occur on the Fiberfrax fume trap or on surfaces within the furnace. In one experiment,(9) the iodine was found condensed in a zone that had temperatures between 450 and 700°C. This indicates that iodine had not volatilized in the elemental form or reacted with other volatile elements in the vapor to form compounds. It is suspected that the iodine volatilized or deposited as cesium iodide, since cesium was present in the temperature zone in excess of the stoichiometric requirement for cesium iodide. However, the total iodine accounted for in this experiment was only fifty percent of that in the charge material. It is believed that the remaining iodine was lost in handling the experimental apparatus.

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* Stack air flow of 60,000 cfm is equivalent to 1700 m^3/min.

The reduction in iodine concentration that occurs by deposition of iodine in the melt refining furnace (by about a factor of 5×10^{-5}) is sufficient, in combination with stack-gas and atmosphere dilution, to reduce iodine concentration to below the maximum permissible level at ground level. A charcoal trap, therefore, constitutes a safety device. For elemental iodine, the charcoal trap should have an iodine-removal efficiency of about 99.9 percent. Particulate iodine activity should be largely removed on AEC filters placed on the fore and aft sides of the charcoal. A charcoal trap is placed within the argon cell and on the suction side of the vacuum pump in order to minimize the shielding requirements for subsequent off-gas handling apparatus (vacuum pump, filters, piping, and the off-gas holdup tank). Nevertheless, 2 ft of concrete shielding has been provided around the vacuum pump and 1 in. of steel shielding has been provided around the holdup tank (largely because of the contained xenon activity). The retention of off gases in the 500-ft^3 holdup tank for about 30 days would result in considerable decay of any iodine activity present.

The holdup tank, which is fabricated of $\frac{1}{4}$-in. steel plate, is divided into four sections by baffles. As an added safety precaution, 1000 lb of activated charcoal is loaded into the first section to trap any elemental iodine which may be present. In the event that an unforeseen buildup of activity occurs in the holdup tank, provision has been made for the addition of 2 ft of concrete shielding around it.

One additional equipment item which has been provided to insure against iodine release in excessive concentrations is a "silver tower" (a tower filled with ceramic pieces on which silver nitrate has been deposited). Operation of the tower at about 200°C removes iodine by reaction of the silver nitrate with iodine to give silver iodide. The silver nitrate tower can be bypassed.

Iodine activity is released from deposited iodine by exposure to air or oxygen. For this reason, release of iodine is expected during the skull oxidation step. The off-gases from this step may be treated, with one exception, in the same manner as are off-gases from the melt refining operation (i.e., passage of the off-gases through a charcoal filter assembly, holdup tank, possibly the silver tower, and disposal from the suspect stack). The exception is that excess oxygen present in the off-gases from the skull oxidation furnace will probably be removed on hot copper in an oxygen-absorption unit. However, at the time of this writing, the method of handling off-gases from the skull oxidation furnace has not been decided.

Iodine may also be released by the direct removal from the argon cell of argon gas which is filtered and exhausted through the suspect stack. Such removals of argon may be necessary for "cell breathing" (i.e., removal of gas to maintain proper pressure differentials) and for the evacuations of transfer locks. The evacuation of locks should remove enough argon to take care of necessary cell breathing.

Steunenberg(84) has estimated that about 0.14 percent of the iodine may be released to the cell atmosphere through nitridation of discharged fuel pins. While the bulk of this material will be absorbed on surfaces, this absorption is ignored in the following calculations. The daily input of iodine to the cell gas would be (for 2 percent burnup, 15-day cooling, and 0.14 percent release to the cell gas) 31.4 c of iodine-131, resulting in an equilibrium level of 365 c or 0.0056 c/ft^3. It has been estimated that in evacuating the locks, about 1000 ft^3 of gas might be removed each week at a rate of 40 cfm (i.e., over a total of 25 min).(80) This is a conservative estimate; the probable normal weekly volume will be about 300 ft^3. If release were permitted in a single 25-min period rather than in shorter increments spread over the entire week, the ground-level concentration could conservatively be permitted to rise to 10 times the maximum permissible concentration (MPC). The permissible iodine discharge rate would be 0.3 c/min. The predicted discharge rate is 0.23 c/min.

Argon may also be released directly to the atmosphere in an emergency. The venting of the cell gas directly to the atmosphere is considered an emergency event, such as might result from failure of the cell gas-cooling system or a sudden extensive drop in atmospheric pressure. The maximum outflow of gas which could occur is about 6000 ft^3 over a period of 6 min. This would result in the discharge of 5.6 c/min and would result in the permissible concentration at ground level being exceeded by a factor of about 200 (assuming no atmospheric dilution). However, since this is an emergency condition, it should be possible to permit such iodine concentrations. The level attained would be only five times that permitted for a discharge of activity occurring over a one-hour period in one week.

The flow of gaseous wastes is summarized in Figure 54.

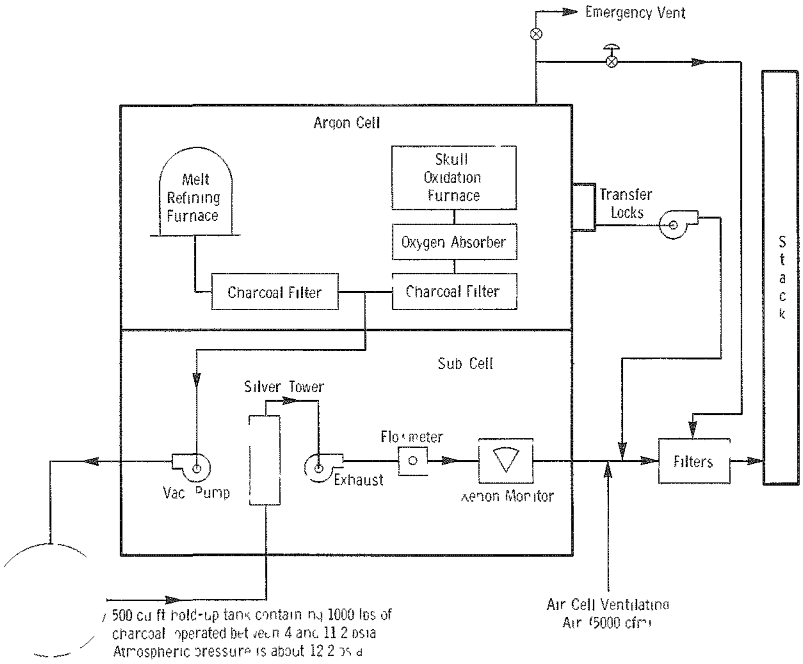

Figure 54

FLOWSHEET FOR DISCHARGE OF GASEOUS WASTES FROM EBR-II FUEL CYCLE

2.7.2 Solid Waste Materials

The solid wastes consist of: scrap materials from the disassembly of subassemblies, fuel-cladding scrap, used melt refining and injecting-casting crucibles, crushed Vycor molds, fume traps, and discarded equipment items. Waste flux and metal from the skull reclamation process will be cast directly into waste containers, not yet designed. The bulk of the radioactivity will, in fact, be disposed of in waste metal and flux streams of the skull reclamation process.

2.7.2.1 Scrap-handling Coffin and Containers

The equipment at the facility for disposal of waste material consists of a scrap-handling coffin and standard disposal containers. The scrap-handling coffin is similar in outside dimensions to the interbuilding (subassembly-transfer) coffin; however, no cooling is provided. The coffin is designed to be used in a vertical position and has a top opening which is closed by a stepped plug. The overall height is about 10 ft; the outside diameter is about 34 in. The storage or center tube is made of stainless steel and is 13 in. in internal diameter and about 7 ft long. The shielding consists of 10 in. of lead at the sides and 8 in. at top and bottom. The coffin is covered by steel sheeting about $\frac{1}{4}$ in. thick. It is loaded while on the transfer cart under the large air-cell floor port, and is moved by the cart and the high-bay-area crane to a truck for hauling to the burial ground. The operation of the cart and crane are the same as for the interbuilding coffin. In addition, another coffin which will hold 55- and 30-gal drums will be available at the site and may be used.

Three standard sizes of disposal containers are planned. They are:

1) 30-gal drums about 16 in. in diameter and 3 ft high;
2) containers, $11\frac{1}{4}$ in. inside diameter and 6 ft high; and
3) containers, about 11 in. in outside diameter and 1 ft high, which will fit inside the 6-ft container described in 2) above.

The containers will have crimp-on lids with asbestos gaskets. The 30-gal drums are planned for large items, such as machinery. The containers will be transferred into the air cell through the large transfer port.

2.7.2.2 Description of Solid Wastes

Disassembly Scrap

The disassembly scrap per core subassembly consists of:

1) about 62 in. of hex tubing;
2) the top fixture with grid and tie bolt;
3) two nuts and two grid plates;
4) the bottom adapter with grid and tie bolt in two pieces.

This scrap is to be placed in the 6-ft containers. It is estimated that one container will hold the scrap from about 10 subassemblies. The stainless steel scrap will have an activity of about 3 c/g.

Cladding Scrap

The fuel-element cladding scrap, which also has an activity of about 3 c/g, consists of the spacer wire, sheared ends, and the decanning scrap ribbon. The ribbon is about $\frac{1}{8}$ in. wide and 5 ft long. For a subassembly of 91 fuel elements, the scrap will amount to about 865 g of wire and ends, and 820 g of ribbon. In addition, there are about 91 g of bonding sodium per subassembly of which an estimated one-half adheres to the pins and one-half (45 g) adheres to the scrap. Thus, the cladding scrap would amount to about 1730 g per core subassembly. The cladding scrap will be placed in the one-foot containers.

Melt Refining Crucibles

The melt refining crucibles of zirconia are $6\frac{3}{8}$ in. in diameter and $9\frac{1}{2}$ in. high. After the skull is oxidized and the skull oxide is dumped into a storage pan, the crucible is released from its holding can and dumped into a one-foot-high container. Some breaking of the crucibles may be desirable to enable more efficient packing.

Melt Refining Fume Traps

The Fiberfrax fume traps are approximately 10 in. in diameter and $5\frac{1}{2}$ in. high. After one use, they will be disposed of in the one-foot-high containers. One container will hold two fume traps plus some other scrap.

Melt Refining Ingot Molds

The graphite ingot molds are $4\frac{15}{16}$ in. square and $6\frac{1}{8}$ in. high. When they become damaged, they will be disposed of in the one-foot containers.

Melt Refining Charcoal Filter Assembly

A decision to use a charcoal filter assembly in the melt refining off-gas line was made just previous to the writing of this report. The design has not been completed. The unit will be installed in the vacuum line to the furnace by freeze-metal joints. It is expected that the unit with connections will be designed so that it will fit into the one-foot disposal containers.

Injection-casting Crucibles

The thoria-coated graphite crucibles are $6\frac{3}{4}$ in. in diameter and $5\frac{3}{4}$ in. high. It is expected that they will be used once and then be disposed of in the one-foot containers.

Vycor Glass from Injection-casting Molds

It is anticipated that from 120 to 160 Vycor molds will be used per casting operation. The broken Vycor from 120 molds will weigh about 2300 g and will be disposed of in the one-foot containers.

Injection-casting Pallets

The pallets or mold holders are 6 in. in diameter and $20\frac{1}{8}$ in. long. It has not been determined whether they will be re-used or not. When disposal is necessary, the 6-ft containers will be used.

Miscellaneous

Miscellaneous fuel-processing items, such as thermocouples, will be disposed of in the 1-ft or 6-ft containers. Small items of processing equipment, such as susceptors and furnaces, will be disposed of in the 30-gal drums. Small items of cell equipment, such as lamps and motors, will likewise be disposed of in the 30-gal drums. Large items will be treated as special problems and individual disposal methods will be used.

2.7.2.3 Movements from Cells

Present plans are to have tools for crimping the lids on the one-foot containers in the argon cell and tools for the 30-gal drums and the 6-ft containers in the air cell. The one-foot containers will be filled and the lids crimped on in the argon cell. They will be transferred to the air cell for loading into the 6-ft containers. Some 30-gal drums will be filled in the argon cell when necessary and transferred through the large lock to the air cell for sealing. Others will be filled in the air cell. Lifting tools with release mechanisms, operated by the bridge manipulators, are to be used for moving the containers and drums.

If excess release of active fumes from melt refining fume traps and filters occurs during evacuation of the large lock, several alternatives can be used. The filters could be sealed before being placed in containers, or the fume traps might be placed in small sealed containers and transferred through the small locks with the use of special transfer devices.

2.7.3 Disposal of Liquid Waste

As pointed out previously, liquid wastes are generated in the process for removing sodium from subassemblies and in analytical operations. The facilities for suspect liquid waste disposal consist of:

One 1700-gal-capacity retention tank in the sodium-disposal system on the service floor of the Fuel Cycle Facility.

Three 1700-gal-capacity suspect retention tanks in the Laboratory and Services Building.

One waste evaporator in the Laboratory and Services Building capable of concentrating 1500 gal (with 2.2 percent solids) to 30 gal (with 40 to 50 percent solids) in 8 hr with a decontamination factor of 10^5 for the evaporated liquid.

Two 150-gal-capacity, high-activity tanks for the analytical caves in the Laboratory and Services Building.

One leaching pit.

One solidification cell for loading liquid from the high-activity tanks into 30-gal drums containing vermiculite and for loading the bottoms from the evaporator into 30-gal drums.

The liquid from the 1700-gal retention tank in the sodium-disposal system is pumped to the three retention tanks in the Laboratory and Services Building. The liquid from these three tanks is pumped to the waste evaporator or to the leaching pit, depending upon the activity level. The overheads from the evaporator are pumped to the leaching pit. The bottoms from the evaporator and the liquid from the high-activity tanks are absorbed in vermiculite and sent to the burial grounds in shielded containers.

A minimum of liquid waste from the Fuel Cycle Facility is expected to be processed by the system. In normal operation, the only activity-laden liquid should be that from the sodium-disposal system.

The activity in this system can arise from impurity elements in the sodium or from the cleaning of ruptured fuel elements. (There is a possibility that fuel elements and repairable equipment that are removed from the argon cell will be cleaned. Cleaning would be by ultrasonic cleaning in a bath of chlorinated hydrocarbons. These liquids would be handled in a closed recycle system with the concentrated contamination being directly absorbed and buried in shielded containers.)

Activity handled in the liquid-waste-handling system is limited because of the unshielded design of the evaporator. Limits for normal operation are expected to be 2 to 5 c for a concentrate of 15 to 25 gal. Limits on the overheads from the evaporator are listed below:

Concentration Limits for Liquid Disposal to Leaching Pit

Isotopes	Total Maximum Concentration ($\mu c/ml$)
Unidentified	10^{-7}
Sr^{90}	10^{-6}
Sr^{90}, I^{129}, I^{131}, $Ru\text{-}Rh^{106}$	10^{-5}
Sr^{90}, I^{129}, I^{131}, Ce^{144}, Sr^{89}, $Ru\text{-}Rh^{106}$, Cs^{137}	10^{-4}
Sr^{90}, I^{129}, I^{131}, Ce^{144}, Sr^{89}, $Ru\text{-}Rh^{106}$, Cs^{137}, Ce^{141}, $Zr\text{-}Nb^{95}$	10^{-3}

If high gross activity levels occur in the sodium-disposal tank, the action taken will be to pump 2- to 5-c amounts to the evaporator for disposal. This disposal will involve dilutions and concentrations tailored to the specific situation. It is also possible to transfer liquids of higher activity directly to shielded containers for burial. Approximately 500 c are the maximum that can be directly transferred to a 30- or 55-gal drum.

2.8 Special Problems

2.8.1 Criticality

Nuclear criticality in the EBR-II Fuel Cycle Facility was the subject of a study by Hesson and Burris.(85) Quantity and configuration limits for EBR-II fuel of the first core composition during storage, handling, and processing operations in the Fuel Cycle Facility were recommended. The recommendations* were based on published nuclear safety guides.(86-94)

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* These recommendations do not yet constitute official policy. They may constitute the basis of an official policy, the establishemnt of which is the responsibility of the operating division. They are included in this report, prior to the adoption of an official policy, to indicate the general nature of the criticality problem.

The philosophy adopted was to establish maximum quantities of fuel materials which could be handled, without risk of nuclear incident, in all fuel-processing operations under the assumed conditions of "full reflection" and with a safety factor of at least two to take care of inadvertent double batching. The assumption of "full reflection" (water at least 3 in. thick or its nuclear equivalent) obviates the need for consideration of the proximity of materials other than fuel or fissionable material. Since "full reflection" is not easily achieved, the assumption that "full reflection" exists provides a considerable safety factor.

It is considered worthwhile to quote, from the above-mentioned study,(85) the recommended allowable limits of fuel, graphite, and beryllia in the EBR-II Fuel Cycle Facility.* The allowable limits were stipulated as maximum-sized unit amounts of fuel, enriched uranium, and plutonium, and restrictions were placed on the geometrical arrangements of the unit amounts.

2.8.1.1 Criticality Control for Core I Fuel and Blanket Materials

The total allowable quantity of fuel materials in a given area is a function of the weight of fuel in "fuel units" and the spacing of the fuel units. Thus, the smaller a fuel unit and the greater the distance between fuel units, the greater is the total allowable quantity of fuel material. This relationship for Core I fuel is shown on Table 15. For a center-to-center

Table 15

ALLOWABLE QUANTITIES OF CORE I FUEL (50 PERCENT U^{235}) AS FUNCTIONS OF SIZE AND SPACING OF FUEL UNITS[a]

	Total Fuel Weight and Number of Fuel Units for Fuel Units Sizes[b] of							
	15 kg		20 kg		25 kg		32 kg	
Center-to-center Spacing of Fuel Units (in.)[a]	Fuel Wt (kg)	No. of Units	Fuel Wt (kg)	No. of Units	Fuel Wt (kg)	No. of Units	Fuel Wt (kg)	No. of Units
12	310	20	250	12	210	8	175	5
15	510	34	410	20	350	14	290	9
18	770	51	620	31	525	21	435	13
21	>1000	>66	860	43	740	29	610	19
24	>1000	>66	>1000	50	1000	40	830	26

[a] It is also necessary that there be 8 in. of clear space between fuel units.

[b] Data adapted from TID-7016, Figure 22, with a factor of safety greater than two.

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* See footnote on previous page.

distance of 24 in. between fuel units with at least 8 in. of intervening clear space, the maximum recommended sizes of Core I fuel units (which permit inadvertent double batching) are the following:

Fissionable Material	Maximum Weight of Unit Quantity (kg)
Core I fuel (46 w/o U^{235})	32 as total fuel
Plutonium (in blanket material at concentrations of less than one percent)	2.6 as Pu^{239}
Fully enriched uranium (93 w/o U^{235})	10 as total U

The recommended limitations on the total quantities of fuel materials present simultaneously in the EBR-II Fuel Cycle Facility are:

Core I fuel (46 w/o U^{235})	500 kg (This corresponds to 16 maximum-size fuel units, i.e., 32 kg of fuel per unit.)
Plutonium (blanket material):	13 kg (This corresponds to 5 maximum-size units, each containing not more than 2.6 kg of plutonium.)
Fully enriched uranium (93 w/o U^{235})	10 kg (Storage in the cell in quantities greater than this should not be necessary.)

The enriched uranium (employed as make-up fissionable material) will be received and, therefore, may be stored in 18-kg-capacity, geometrically safe shipping containers. The 10-kg limitation is, therefore, waived when this material is received and stored in geometrically safe containers, but should be adhered to in subsequent handling. A maximum limit of 200 g for the total of all fissionable materials in all of the liquid-waste-holdup tanks is recommended.

The requirement that there be at least 8 in. of open or clear space between fuel units permits a thick, close-fitting reflector about a cubic array (see Figure 22, p. 26 of TID-7016, rev. 1). In TID-7016, p. 26, it is stated that this spacing criterion is also met if the units are separated by a layer of water or concrete at least 8 in. in thickness.

The amount of potential moderators, graphite (or carbon) and beryllia, should be limited to an accumulation in each cell of 600 kg of graphite and 200 kg of beryllia.

By use of Table 15, p. 172, alternate choices of unit fuel quantities and spacing of fuel units can be made. The nature of the fuel-recovery operations may be such that unit fuel quantities smaller than the permissible maximum may be preferable. Any unit quantity of fuel which is selected can, in turn, consist of a group of smaller quantities of fuel.

For storage of fuel, the Nuclear Safety Guide, TID-7016, permits a relaxation of the customary double-batching safety factors on the assumption that control of individual fuel units in storage is more stringent than in process operations. The minimum spacing for 32-kg fuel units in storage is 12 in. between centers with not less than 8 in. of clear space between units. The spacing of storage pits in the argon and air cells is such that these minimum required distances are in all cases exceeded. Even though a 12-in. spacing of stored 32-kg fuel units is adequate, it is recommended that a uniform set of rules be adopted for both storage and handling of fuel in the cells, namely, 2 ft between adjacent fuel units with 8 in. of intervening space.

Although the 32-kg limitation also applies to the storage of melt refining skulls or skull oxide in a storage pit, additional precautions must be taken in the storage of these materials because of the remote possibility that, through water inleakage, the pits may flood with water. It is recommended that the skull oxide be stored in geometrical arrangements which provide a safe condition even though the containers and pits fill with water. Two recommended arrangements are: (1) containers with safe diameters, i.e., less than 5.5 in. in diameter, and (2) containers 8 in. in diameter and 4 in. high with at least 10 in. of clear space between containers. The cans may fill with water, but oxide cannot be permitted to escape from the can. Because the oxides will have a density of around 10 g/cm^3, movement of oxide even in an open can filled with water would be unlikely.

Fuel subassemblies awaiting processing or return to the reactor may be stored in storage pits in the Argon Cell. Under the worst-assumed conditions, i.e., water flooding of the pits, at least seven subassemblies would be required for a critical mass. Therefore, not more than four fuel subassemblies should be stored in a storage pit.

Mechanical aids and administrative control will be required to ensure that unit quantities are not exceeded and that geometrically safe configurations are maintained. The following mechanical aids are recommended for maintaining the spacings required for geometrically safe configurations and for preventing an accumulation of fuel in amounts in excess of the maximum recommended quantities:

1) Mechanical apparatus to limit the maximum number of assemblies in any one air cell storage pit to four (4) subassemblies.

2) Mechanical apparatus to limit storage of oxide or fuel in the argon cell storage pits to geometrically safe configurations described above.

3) In the case that fuel elements are cleaned (as has been occasionally proposed) in a halogenated carbon fluid (such as trichloroethylene), the cleaning container should have geometrically safe dimensions (less than $5\frac{1}{2}$ in. in diameter). The halogenated carbon fluid would not be an effective moderator, but there is the possibility of inadvertent substitution of water.

The recommended limits of fuel (quantity and configuration) given in the preceding sections should impose no severe operational problems, since the weights of fuel in individual major fuel items or in various configurations are, as shown below, much less than the recommended limit of 32 kg per fuel unit:

Item	Fuel Weight, g
One subassembly	6,070
One fuel element	67
One fuel pin	67
Melt refining charge	about 10,000
Melt refining ingot	about 9,000
Melt refining skull	about 1,000
Injection-casting charge	about 14,000
Injection-casting heels	about 4,000
Fuel-pin castings	about 100
Scrap	variable

2.8.1.2 Criticality Considerations for Future Core Loadings

It is anticipated that subsequent cores with plutonium as the fissionable material in depleted uranium will be fabricated. In such cases, the percentage of plutonium in the fuel will be less than the percentage of uranium-235 in the current fuel because of the greater reactivity of plutonium. If the size of the plutonium fuel core is the same as that of the present core, the allowable mass for a unit of unmoderated but reflected fuel would be about the same as in the current case (i.e., about 32 kg per fuel unit) since the decreased concentration of the plutonium would compensate for its greater reactivity.(87) However, in cases of possible effective moderation of the fuel, as by water, or in the event of accidental fuel enrichment, the minimum critical mass would be considerably less.

In the case in which a larger reactor core might be used with, say, only 25 percent uranium-235 enrichment in the fuel, larger unit masses could be used. A Nuclear Safety Guide(86) shows an allowance factor of 2.5 for 25 percent enrichment. The "fully reflected" minimum critical mass would be 228 kg of fuel (22.8 x 2.5 x 4), and the recommended unit weight would be 100 kg of fuel (10 x 2.5 x 4).

In the case of future cores, mixtures of plutonium and uranium-235 in natural uranium might be used as fuel. The following tabulation(87) shows the critical masses of uranium-235 and plutonium for various volume fractions in uranium-238 with a thick ($7\frac{1}{2}$-in.) uranium reflector. The volume fraction of uranium-235 is based on a density of 18.8 g/cm^3 and that of plutonium is based on a density of 14.2 g/cm^3.

Percent (Volume) in U^{238}	Minimum Critical Mass, kg	
	U^{235}	Plutonium
100	15	6.5
90	16.5	7.1
80	18	7.8
70	19.5	8.7
60	22	9.9
50	25	11.4
40	30	13.7
30	36	17.2
20	50	24

Assuming that 1 g Pu^{239} is equivalent to 3 g U^{235}, a fuel with 20 v/o (16 w/o) plutonium and 20 v/o (21 w/o) U^{235} would have a total critical mass of 45.6 kg of fuel (7.2 kg Pu^{239} and 9.9 kg U^{235}). For a full reflector corresponding to about 2 in. of uranium (or infinitely thick water), the critical mass would be about 65 kg of fuel, which is about the same as the critical mass, 72 kg, for the 50 percent U^{235} fuel (Core I fuel).

On the same basis, a fuel of 35 v/o (29 w/o) plutonium in natural uranium would have a minimum critical mass of about 76 kg of fuel for a full reflector, while a fuel of 25 v/o (20 w/o) plutonium would have a minimum critical mass of about 115 kg of fuel.

2.8.2 Accountability

From the viewpoint of overall accountability in the Fuel Cycle Facility, the facility might be considered from the "black box" viewpoint. There will be incoming, accumulation of, and outgoing fuel, uranium and plutonium. The accumulation will be merely the difference between incoming and outgoing material. These items would consist of components as follows:

a. Incoming Fuel

    1) From reactor
    2) Makeup

b. Fuel Accumulation

1) Process inventories
2) Skull or skull oxide

c. Outgoing Fuel

1) To reactor
2) Unrecoverable elements or spill material
3) Losses in cladding, crucible, and mold scrap
4) Waste streams from skull reclamation process
5) Analytical and other samples.

The "black box" concept requires accurate knowledge of all incoming and outgoing fuel items. Items (1) and (2) under both incoming and outgoing fuel would probably be known with sufficient accuracy for the concept to be useful. However, the average magnitudes of the outgoing losses in cladding, crucible, and mold scrap (item c-3) must be determined. This will be especially important in order to prevent accumulation of a critical mass.

Since the reactor fuel contains uranium-235 and since plutonium will be bred in the fuel and blanket materials, accountability of fuel and blanket materials will be involved. Thoria is used as a coating for injection-casting crucibles and molds. Accountability of this material may also be required. Inventories of these materials will be as follows.

2.8.2.1 New Makeup Uranium and Fuel Alloys

The accountability of the natural uranium, uranium enriched in uranium-235, fissium alloy, and fissium alloy in new fuel elements can be handled with standard procedures. As the material is transferred into the cells, accountability can be transferred to the fuel cycle.

2.8.2.2 Fuel in Fuel Elements in Subassemblies in the Reactor and Coffin

All subassemblies will be numbered, and individual fuel elements will each occupy a numbered position in the subassembly. Fuel in subassemblies will be identifiable as numbered fuel elements in a numbered subassembly. Records will show the locations of the subassemblies. From processing or manufacturing information, from analyses, and from reactor operation information, the history, analysis, and weight of the fuel pin in each fuel element will be known. As the subassemblies are transferred to and from the reactor and Fuel Cycle Facility, accountability can be transferred accordingly.

2.8.2.3 Fuel Being Reprocessed in the Fuel Cycle Facility

After the fuel has been exposed in the reactor, it will be too active to be stored outside the cells or coffins. Hence, accountability will need to be maintained on a remote basis.

2.8.2.3.1 Identification of Fuel Items

Since many items of fuel, such as fuel elements, ingots, and pins, may not be numbered or marked directly on the item, identification may be by position. Fuel-element magazines, oxide- or skull-storage cans, storage locations, etc., can be numbered for identification.

After disassembly of a subassembly and until decanning and chopping, each fuel element can be individually identified by giving it the numbered position it occupied in a specific subassembly and keeping a record of the position it occupies in a magazine.

After decanning and chopping the fuel in the fuel pins (except for samples) becomes part of a melt refining charge batch and loses individual identity. The melt refining ingot and the skull can be identifiable with the melt refining batch number.

The melt refining ingots, remelt ingots, and possibly the heels become part of an injection-casting batch number. The history of the charged material will be recorded. The fuel pins, heel, and scrap from the injection-casting batch can be identified with the injection-casting batch number.

If heels and scrap are remelted, the remelted batch will have a number and the history of the charged material will be known. The remelt ingot and skull or residue from the remelt can be identified with the remelt batch number.

The fuel pins and fuel elements from a given injection-casting batch can be identified with the batch number. They will be assigned individual identity as they are measured and weighed. After the fuel elements are reassembled in a subassembly, they can assume the identity of a numbered position in a specific subassembly.

Records of the fuel could thus contain a complete history of any item of fuel on an operation-to-operation basis.

2.8.2.3.2 Quantities of Uranium and Plutonium in Inventories of Fuel

The quantity of uranium and plutonium in any fuel item will be a product of the weight of the item and the fraction of fissile material present. Scales or balances will be used for weighing, and chemical analyses will be made. In the cases where analyses are not made, estimates based on previous records can be made.

The scales planned for the argon cell are as follows:

Range	Estimated Accuracy, g	Remarks
5- 25kg	±20	Toledo platform-dial type
4-20 kg	±2	Servo-recording mechanism
0-4 kg	±1	Differential transmitter type
67-71 g	±0.1	Differential transmitter type for pin processing.

There will be some uncertainty in the quantities of uranium and plutonium in the fuel inventories due to inaccuracies in analyses, estimates, and weights. The quantities of uranium and plutonium in the melt refining skulls or skull oxide may be among the most uncertain, because of the difficulties in obtaining representative samples for analyses. It may be more accurate to use the difference in uranium content between the charge and ingot as the weight of the skull.

2.8.2.3.3 Fuel Losses

Fuel losses may occur as follows:

a) with cladding scrap;

b) with melt refining, remelting, or injection-casting crucibles;

c) with melt refining ingot molds or injection-casting Vycor pin molds;

d) as badly damaged fuel elements not suitable for reprocessing;

e) as fuel from a spill or leak;

f) in waste streams from the skull reclamation process.

The quantities of uranium and plutonium lost will be difficult to determine accurately because of the difficulty of obtaining representative samples. Since cladding scrap, used crucibles, and molds will be sampled only occasionally for uranium losses, it may be necessary to establish values to be used for average losses.

2.8.2.4 Blanket Material

The accountability of the blanket material will present problems somewhat similar to those involving the fuel. The blanket material originally consists of depleted uranium. After it has been exposed in the reactor, it will be too active to be exposed outside the cells or coffins. Hence, accountability will need to be maintained on a remote basis. The blanket elements can be identified in a manner similar to fuel elements.

2.8.2.5 Thoria Coatings

The injection-casting graphite crucibles and Vycor molds are coated with a thin layer of thorium oxide. Since only about 35 g of thoria are used per crucible and 60 g per 120 Vycor molds, the thoria can probably be charged off as it is used

2.8.3 Radiation Shielding(12)

The air cell and argon cell are designed for radiation levels up to 10^6r/hr and activities up to 5×10^5 c At the operating level the cell walls are 5 ft thick and are made of 220-lb/ft^3 concrete. The roof has a minimum thickness of 4 ft and is made of standard concrete. The windows have a total thickness of 57 in. and are provided with steel radiation shields, about 6 in thick, for protection of the windows from radiation. Electric motors (except on some process equipment) within the cell are shielded. Steps, such as elimination of greases and organic insulation, are taken to reduce the effects of radiation damage.

In order to check the radiation levels in the cells, radiation dosimeters based on the darkening of glass due to exposure to gamma radiation have been developed. The dosimeters consist of glass strips in standard 2-in.-long by $\frac{3}{4}$-in.-diameter, aluminum sample cans It has been found that 0.12-in.-thick glass, Pittsburgh Plate Glass - 6788, can be used for total exposures of about 10^6 to 10^9 r at exposure rates of 3×10^4 to 3×10^6 r/hr.

Although the cells are designed for high levels of radiation, some of the equipment will have its life shortened by radiation damage. It might be possible to reduce the radiation within the cell substantially during nonoperation periods (at night and over weekends) by relatively simple shielding. For instance, fuel elements in magazines and baskets may constitute a major amount of the fuel inventory. The horizontal magazines will be kept with their carts. These could be shielded by movable U-shaped covers of lead. Lead thicknesses of $\frac{1}{2}$ to 1 in. would give substantial shielding. The A, B, and loading magazines could be covered by lead bell shields. The shields could be provided with ventilating louvres to allow for convection cooling.

2.8.4 Unrecoverable Fuel Other than Normal Waste

Normally, fuel would be completely reprocessed and no fuel except possibly skull material would be sent as such from the cells. However, it is possible that unrecoverable fuel material may be generated which will require disposal. Such material might be as follows:

1) subassemblies with overheated or badly ruptured fuel elements;
2) badly damaged or bent fuel elements;
3) fuel from a spill or leakage from either furnace crucible;
4) badly nitrided pins, scrap, or spill material not considered satisfactory for oxidation to skull oxide.

The method of disposal of fuel items becomes a special problem, mainly because of high activity, self-heating, and accountability. Whether an item would be sent to the aqueous chemical processing plant at the National Reactor Testing Station for uranium recovery or would be buried would depend upon the amount and condition of the fuel.

2.8.5 Disposal of Core-subassembly Blanket Element

Thirty-six blanket elements will be obtained from each core subassembly. Each blanket element originally contains two depleted uranium pins, 0.3165 in. in diameter by 9 in. long. The 36 elements contain about 16 kg of uranium. The plutonium content of the blanket pins will be low because of the relatively short time in the reactor. There are no initial plans to re-use the blanket elements. Since the plutonium content of the pins will be only about 0.2 percent at the end of the elements next to the core and will quickly diminish to essentially zero at the other end of the elements, these blanket elements may be buried rather than sent to a plutonium recovery plant. Some blanket elements will be analyzed.

After initial test work, blanket elements might be eliminated from the core subassembly, as is done in the case of control subassemblies. A second alternative is a redesign of the core subassembly so that the axial blanket rods can be reused.

2.8.6 Nitridation of Exposed Fuel(55)

The extent of the nitridation problem for irradiated fissium is at present not fully known. However, tests have been conducted by LaPlante on unirradiated fissium in the form of sodium-coated and uncoated pins and on irradiated fissium pins.(55) Table 16 shows calculated nitridation values for unirradiated pins based on LaPlante's data. Since the rate of nitridation is proportional to the square root of time, the quantities would be ten times as large for 10,000 min (7 days) than are given in the table. A time of 100 min was chosen for the table, since it is approximately the time to decan and charge a batch of pins. Other of LaPlante's data not reported here indicate that the nitridation rates for argon atmospheres containing 0.20, 0.28, 2.04, and 100 percent nitrogen are about 50, 75, 100, and 100 percent, respectively, of the rate for a 5 percent nitrogen atmosphere.

Table 16

CALCULATED NITRIDATION OF UNIRRADIATED FISSIUM(55) PINS AND INGOTS IN ARGON-5 PERCENT NITROGEN ATMOSPHERE

	Temperature, C		
	197	308	504
Rate constant K x 10^{11} [(g/sq cm)^2/(min)]			
Uncoated pins	0.236	5.60	186
Sodium-coated pins	0.958	15.2	325
Grams nitrogen reacted per sq cm in 100 min x 10^5			
Uncoated pins	1.53	7.5	43
Sodium-coated pins	3.06	15.3	57
Nitride thickness in 100 min, mils[a]			
Uncoated pins	0.008	0.039	0.224
Sodium-coated pins	0.016	0.099	0.296
Percent of total uranium nitrided in 100 min			
Uncoated pins (0.144-in. diameter)	0.011	0.053	0.304
Sodium-coated pins	0.021	0.108	0.403
6-kg ingots (square)	0.002	0.011	0.040

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[a] Based on the assumption that the nitride is U_2N_3 (density = 11.24 g/cc)

Table 17 summarizes the results of LaPlante's experiments with irradiated pins. It will be noted that the rates for irradiated pins are similar to those for unirradiated pins. For nitrogen percentages above 1 percent, the irradiated pins would self-heat or ignite at temperatures of about 650 to 730 C. The ignitions occurred near the temperatures when pin swelling and fission gas release started. Estimated temperatures of bare fuel with 2 percent burnup and 15-day cooling with no forced-convection heat removal are:

Decanned pins in crucible charger	350 C	(640 F)
Pins in melt refining crucible at the end of one hour*	650 C	(1200 F)
Ingots	525 C	(975 F)
Cast pin in Vycor molds	393 C	(740 F)
Cast pins in single layer in tray	170 C	(336 F)

Table 17

SUMMARY OF NITRIDATION TESTS OF IRRADIATED 5-w/o,(55) 10-PERCENT ENRICHED FISSIUM PINS

Percent Burnup	Cooling Time (months)	Percent Nitrogen in Argon	Temperature (C)	Grams Nitrogen Absorbed Per Sq Cm Per 100 Min
0	-	2, 5, and 100	300	7×10^{-5}
0.74	8	2, 5, and 100	300	8.5×10^{-5}
1.0	4	2, 5, and 100	300	9×10^{-5}
1.0	$3\frac{1}{2}$	100	300	12×10^{-5}
1.0	$3\frac{1}{2}$	100	450	19×10^{-5}
1.0	5	5	650	620×10^{-5}
1.12	$3\frac{1}{3}$	100	500	57×10^{-5} [a]
1.18	$3\frac{1}{3}$	100	650	800×10^{-5} [b]
1.18	$3\frac{1}{3}$	100	660 [c]	100% [d]
1.18	$3\frac{1}{3}$	5	660 [c]	5% [d]
1.18	$3\frac{1}{3}$	1	730 [c]	1% [d]
1.18	$3\frac{1}{3}$	0.5	None [c]	$\frac{1}{2}$ % [d]

[a] Pin was sodium coated.

[b] in 30 min.

[c] Temperature at which initial self-heating (ignition) of pin occurred at furnace heating rate of 10 C/min and in a flowing nitrogen-in-argon stream.

[d] Estimated extent of pin nitridation, which was limited at low nitrogen concentrations by lack of nitrogen.

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* Estimated minimum time to replace cover and freeze seal before purging.

It seems that nitridation could be a problem which might require use of one or more of the following:

1) Reduce the nitrogen in the argon cell atmosphere much below the nominal estimated 5 percent.
2) Provide argon cooling wherever feasible, as for the pins during decanning.
3) Provide containers with very low nitrogen content for overnight, weekend, and storage for longer term - they might consist of small bell jars and a source of high-purity argon.
4) Accelerate purging of melt refining furnace.

If skulls are stored, nitridation would probably occur with some possible release of iodine. The rate, however, would be low.

In his tests with irradiated pins, LaPlante found that exposure of the pins to normal atmosphere (resulting in the formation of an oxide coating) before heating for nitridation tests reduced the rate of nitridation markedly. This observation, however, has little pertinence to operations in the argon atmosphere of the Argon Cell.

The effects of storing unirradiated pins for 2 hr in argon atmospheres with up to about 5 percent nitrogen at 350 C and 650 C on subsequent melt refining-pour yields were investigated by Bennett.(40) The yields were not appreciably affected by storage under these conditions.

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