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Reactor with very low fission product inventory

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Reactor with very low fission product inventory

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Swiss Federal Institute
of Reactor Technology

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Abstract.

A fast converter with one zone and an internal breeding ratio of 1.00, with liquid fuel in the form of molten plutonium - - uranium - and sodium chloride, with a thermal power of 3 GW (th) allows continuous extraction of the volatile fission products (Br, I, Kr, Xe, Te) by means of helium purging in the core. The non-volatile fission products e.g. Sr and Cs can continuously be extracted in a chemical reprocessing plant at the reactor site. The impact on an accidental release of fission products is rather significant; the amounts released are 50-100 times smaller than those in a reference reactor (LWR with oxide fuel). Because the heat sink is relatively large and after heat reduced, the temperature of the fuel does not exceed 500°C after an accident, which greatly reduces the consequences of an accident.

Zusammenfassung.

Ein schneller konverter Reaktor mit einer Zone und einem Brutfaktor von 1.00, mit flüssigem Brennstoff in Form von geschmolzenem Plutonium -, Uranium - und Natriumchlorid, mit einer Leistung von 3 GW (th) erlaubt die kontinuierliche Enfernung von flüchtigen Spaltprodukten (Br, I, Kr, Xe, Te) durch Durchleiten von Helium im Core. Die nicht-flüchtigen Spaltprodukte wie Sr und Cs werden kontinuierlich in der angegliederten Aufbereitungsanlage ausgeschieden.

Die Folgen bei einer unfallbedingten Freigabe von Spaltprodukten sind recht signifikant; die freigegebenen Mengen sind 50-100 mal kleiner als die beim Referenzreaktor (einem LWR mit Oxiden als Brennstoff). Weil die Wärmesenke relativ gross ist, und die Nachwärme reduziert ist, übersteigt die Temperatur des Brennstoffs nach einem Unfall 500°C nicht, was zu einer bedeutenden Reduktion der Unfallfolgen führt.

INIS DESCRIPTORS

MOLTEN SALT REACTORS
FISSION PRODUCT RELEASE
AFTER HEAT REMOVAL
REACTOR SAFETY
LOSS OF COOLANT
MELT DOWN

Contents.

1. INTRODUCTION

1.1 Present position

The light water reactor, the most common power reactor today and for the next two decades is safe enough to be the basic energy source for society.

This however does not mean that the search for improved safety is a waste of time. The continued search for still safer systems is common in all the technologies making up our civilization (transportation, chemical technology, domestic fire hazards etc.).

Each type of nuclear reactor also has potential for increasing its safe operation. It must however be remembered that the most important safety aspect of the current reactor systems is in the real practical experience built up over many years. Here the light water reactor is in a privileged position being able to demonstrate an excellent safety record.

In spite of this crucial fact the search must continue for reactypes which promise improved inherent safety on the basis of their different system design. Such a search seems to be generally desirable.

The best reason for such a reactor type has been given by Alvin Weinberg (1979, 1980).

> "For the 15 billion curies contained in a 1000 megawatt reactor we could never say that the chance of a serious accidental release was zero.
>
> ... Thus a most important technical fix for nuclear energy would be a means of minimising the amount of land that conceivably could be contaminated in the worst possible accident.

> ... For nuclear energy to survive we must reduce the probability of any serious malfunction much below the 1 in 20 000 per reactor-year estimated in the Rasmussen report as well as reducing any possible consequences."
>
> "But if the world energy system involved as many as 5000 reactors - that is, 10 times as many as are now either in operation or under construction - one might expect an accident that released sizeable amounts of radioactivity every four years. Considering that a nuclear accident anywhere is a nuclear accident everywhere I believe this accident probability is unacceptable. If a man is to live with fission over the long term he must reduce the *a priori* probability of accident by a large factor - say 100."

(The Bulletin of Atomic Scientists.
March 1980)

The aim of this paper is to try and present a concept for a fission power reactor being approximately 100 times less hazardous than existing reactor types, and with much less than 15 billion curies activity.

It must be stressed that this study is concerned with a 'paper reactor' (which is of course the safest type of all!) a concept only. The route from the concept to realising an actual power reactor in service, and contributing to our energy problems is a long and tortuous road and with financial commitments of tens of billions of dollars. Even this should not prevent the search for such a reactor. The search itself is of benefit even for a better understanding and possible improvement to the existing power reactor systems.

How is it possible to meet the requirement for a 100-fold improvement in reactor safety?

The following proposals are made below:

a) the amount of fission products in the core during a normal operation must be reduced by a factor 100 which significantly reduces the effects of a large accident where the fission products would be released into the environment,

b) the decay heat level must be reduced and the internal inherent heat sink, such as the total heat capacity of the fuel and other components must be significantly increased. Additionally the maximum temperature reached by the fuel in the case of failure of all emergency cooling systems must be low enough to allow the fuel to be trapped in a core catcher.

Table 1.1 Reactor characteristics affecting safety
(data for 3 GW(th))

	Characteristics	Existing Reactor (e.g. LWR)	Desired Reactor
1)	Fission product inventory	~15 Gcurie. Possibility of releasing into the environment: a) volatile Fission Products (F.P.) b) non-volatile F. P.	Minimum in F.P. inventory but at least two orders of magnitude smaller than in existing reactors for both: a) volatile F.P. b) non-volatile F.P.
2)	Decay heat removal	Immediately after shutdown ~180 MW(th). The integrated decay heat over some hours is of the order of 1000 Gigajoules	Minimum decay heat. At least by more than one order of magnitude
3)	Pressure in fuel and coolant (in containment)	a)~150 bar in a PWR ~80 bar in a BWR in the coolant. b) internal pressure in the fuel pins	No pressure: a) in the coolant b) in the fuel
4)	Coolant with low boiling point (Explosion possibility)	The presence of water in the primary circuit gives rise to the hazard of uncontrolled boiling and production of large volumes of steam, pressurizing the containment	No low boiling point media allowed in the reactor circuits
5)	Chemically active medium (coolant)	In a liquid metal cooled reactor an exothermic reaction is possible $4\ Na_{met} + UO_2 \rightarrow 2Na_2O + U_{met}$	No chemically active agent is allowed
6)	Hydrogen Evolution	In Light Water Reactors during an accident the following reaction may occur: $Zr_{met} + H_2O \rightarrow ZrO + H_{2gas}$ The evolved hydrogen results in an uncontrolled increase of pressure in the core vessel or containment because of burning in air	No chemical agent including hydrogen is allowed
7)	'China Syndrom'	For the LWR system the report WASH-1400 discussed the probability of the'China-syndrom'	Elimination of this kind of accident
8)	Criticality control	A strongly negative criticality, for the case of loss of coolant : control rods	A self regulating system is desired
9)	External (away from reactor) movement of plutonium	The LWR system being a producer of plutonium results in transporting plutonium away from reactor (e.g. in irradiated fuel)	A self sufficient reactor with a breeding ratio of ~1 has no external circulation of Pu

Both of these criteria seem to be possible in a reactor having a molten fuel with continuous extraction of fission products in the fuel during normal operation. (Ref. 19).

In this paper one solution to this problem is discussed in detail: the molten salt reactor. It must be said that such reactor types have been considered for many years having molten salt as a fuel and with continuous extraction of fission products.

Some examples of molten salt reactors: (Ref. 5, 14, 16, 17, 18).

a) thermal breeder with molten fluorides developed decades ago by Oak Ridge National Laboratory, existing as an experimental reactor with a power of 8 MW(th), (Ref. 13)

b) fast breeder reactor with molten chlorides discussed for tens of years, but still a "paper reactor".

A reactor of the second type is discussed here.

1.2 Safety of a Fission Reactor: state of the art.

Of course the hazard of a fission reactor is not only connected with its fission product inventory. There are other important safety aspects. Table 1.1 shows the impact of other reactor parameters on its safety and ways of resolving these. The question arising out of these criteria is can all the necessary improvements be made in a single reactor type? The answer assumed here is yes.

1.3 Safety of a Fission Reactor: possible improvements.

Table 1.1 summarises very briefly the properties of such a 'super safe' reactor. The reactor proposed in this paper is called SOFT.

S alt reactor
O n site reprocessing
F ast converter
T ask

The properties are discussed below in the order given in table 1.1.

Table 1.2 Safety of a Fission Reactor: possible improvements

Parameter	Properties of the desired reactor	The solution formed for the SOFT reactor
1) Fission product inventory	Minimum F.P. Two orders of magnitude less volatile and non-volatile F.P.	Continuous extraction of: a) volatile F.P. by means of gas pumping b) non-volatile F.P. by means of chemical treatment and storage in another containment
2) Decay heat removal	Decay heat at least one order of magnitude less	Decrease of decay heat energy by: a) continuous extraction of F.P. b) continuous extraction of ^{239}Np and storage in another containment
3) Pressure	Works at ambient presssure a) in coolant b) in fuel	Use of molten salt as both: a) the coolant b) the fuel in a pressureless system
4) Low boiling point media (Explosion possibility)	Absence of agents having low boiling point reducing also the explosion hazard	Both fuel and coolant have a boiling temperature of approx 1500°C thus no explosion is possible
5) Chemically active media	Absence of chemically active media which could react with fuel e.g. sodium and oxide	Both fuel and coolant are in the form of molten chlorides. The thermodynamic stability of all constituents exclude any exothermic chemical (or explosive) reaction
6) Hydrogen evolution	Absence of substances which could give rise to hydrogen. E.g. water reacting with Zirconium	Neither fuel nor coolant contain hydrogen
7) 'China Syndrome'	Elimination of any formation of a mass of molten fuel, slumping through containment	The significant decrease in decay heat and the presence of large heat sinks (reflector, core catcher) eliminates a 'China Syndrome'
8) Criticality control	Fuel inherently 'self controlling'	Fluid fuel reactors are known to have a large negative temperature coefficient and Doppler coefficient

1.4 The present State of Reactor Development and the Criteria for a Safer Reactor

The present state of reactor technology results from a long and complicated development and marked by not always the most logical decisions.

If the following are the four main targets of reactor development:

a) an economically and technically feasible source of electrical energy,
b) economically and technically feasible source of high temperature heat for chemical processes,
c) high grade utilisation of uranium resources (and even thorium resources),
d) satisfactory level of safety,

then it may be possible to show with the aid of a diagram (Fig. 1.1) the present state of reactor development.

Of course there are other ways of eliminating the characteristics presenting the greatest hazards - high pressure and high F.P. inventory. Figure 2 shows some possibilities. It seems however that only molten salts can solve both these problems.

In this paper the accent is on the safety problem i.e. the effort to reduce the consequences of the worst credible accident - core melting. Paradoxically as this might sound the solution to this problem is the use of a reactor with a molten fuel core.

In this reactor:

a) the most important parameter to be optimised is the reduction in the inventory of volatile fission products (which will control the consequences of an accident during the first hours) and fission products such as Sr-90 and Cs-137 (which determine the accessibility of the contaminated area for tens of years),

Fig. 1.1

TRENDS IN THE PRESENT DEVELOPMENT OF THE REACTOR TECHNOLOGY

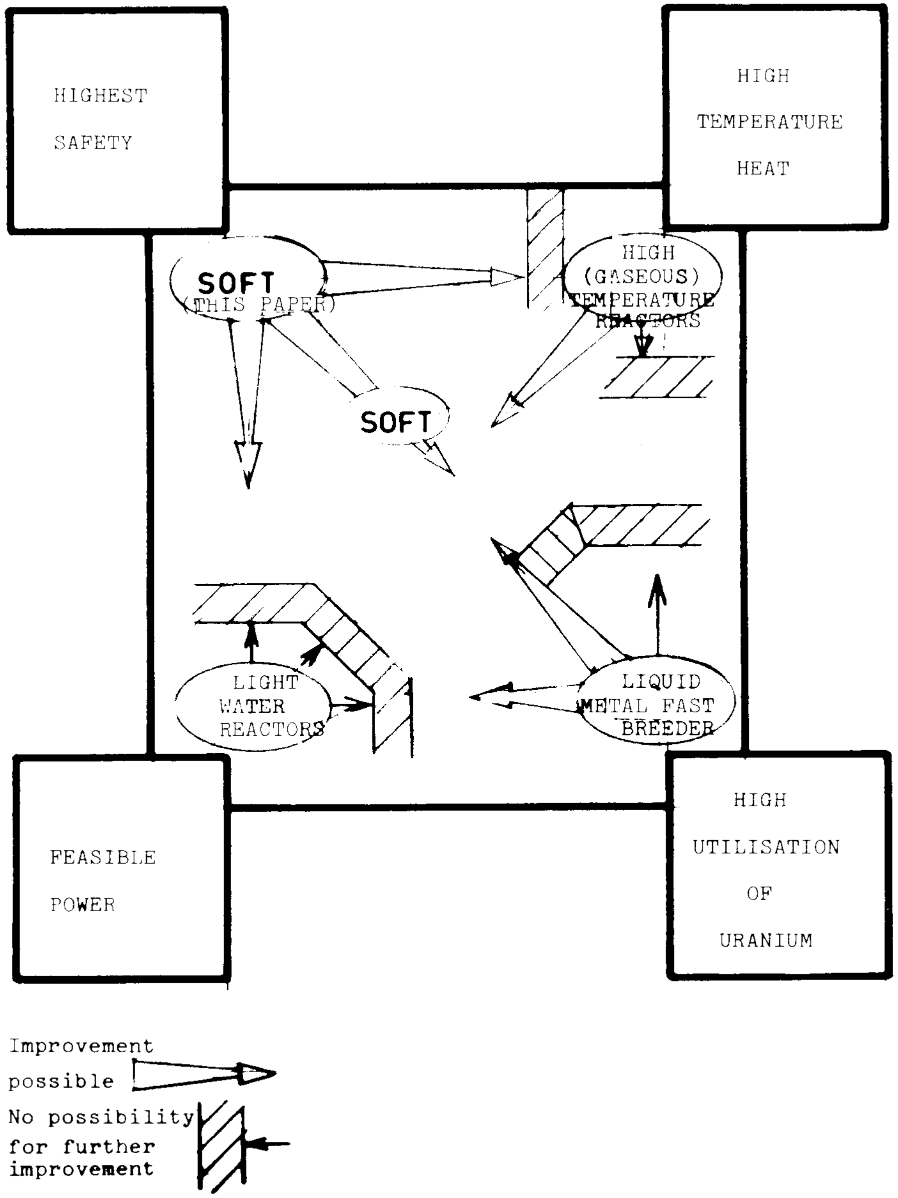


Fig. 1.2

DIFFERENT REACTOR TYPES AND THE SAFETY PROBLEMS

		Pressure in the coolant or fuel	
		No (good : +)	High (bad : -)
Fission product inventory and after-heat	High (bad : -)	Liquid Metal Fast Breeder (+)(-)	Light Water Reactor (-)(-)
	Low (good : +)	Molten Salt Reactor (thermal-fluoride; fast-chloride) (+)(+)	Gaseous Fuel Reactor (-)(+)

b) decay heat resulting from the spontaneous radioactive decay of fission products and actinides but also of the structural material is significantly lower, even if it is not enough,

c) the breeding ratio has been set at 1 in order to minimise the need to transport fissionable material. this makes the reactor a net zero consumer and zero producer of fissionable material,

d) the entire system fuel, primary and secondary cooling systems have been designed to operate at ambient pressure,

e) no highly reactive compounds, (e.g. metallic sodium) are present in the system,

f) no hydrogen containing compounds (e.g. water), which can evolve free hydrogen, are present. (see Ref. 14, 15, 16, 17, 18).

2. ACCIDENTAL RELEASE OF FISSION PRODUCTS AND DECAY HEAT REMOVAL

2.1 The Rasmussen Scenario (Ref. 12, 24).

The scenario of an accident used here is taken from the report WASH-1400 as the most fatal case: the PWR-1 which can be characterised by a steam explosion on contact of molten fuel with water in the reactor vessel. This accident category includes the following fraction of fission product core inventory release:

	Rasmussen	German risk study (maximum)
Xe - Kr	0.9	1.0
I	0.7	0.8
Cs-Rb	0.5	0.5
Te-Sb	0.4	0.35
Ba-Sr	0.05	0.052
Ru (Rh,Co,Mo,Tc)	0.4	0.38
La (Y,Zr,Nb,Ce,Pr, Nd, Np, Pu, Am, Cm)	3×10^{-3}	2.6×10^{-3}

The release of fission products and actinides to the environment results in two different scenarios (simplified here):

a) the impact of volatile short lived fission products from the passing cloud in the direct neighbourhood of the reactor - some hours after the accident,

b) the impact of non-volatile long lived fission products.

2.2 The Problem

From consideration of this scenario of the worst reactor accident it is clear that to improve the safety significantly the proposed reactor must allow the continuous extraction of both classes of fission products - the short lived volatile F.P. and the long lived non-volatile F.P. This is the most important aspect. (Table 2.1)

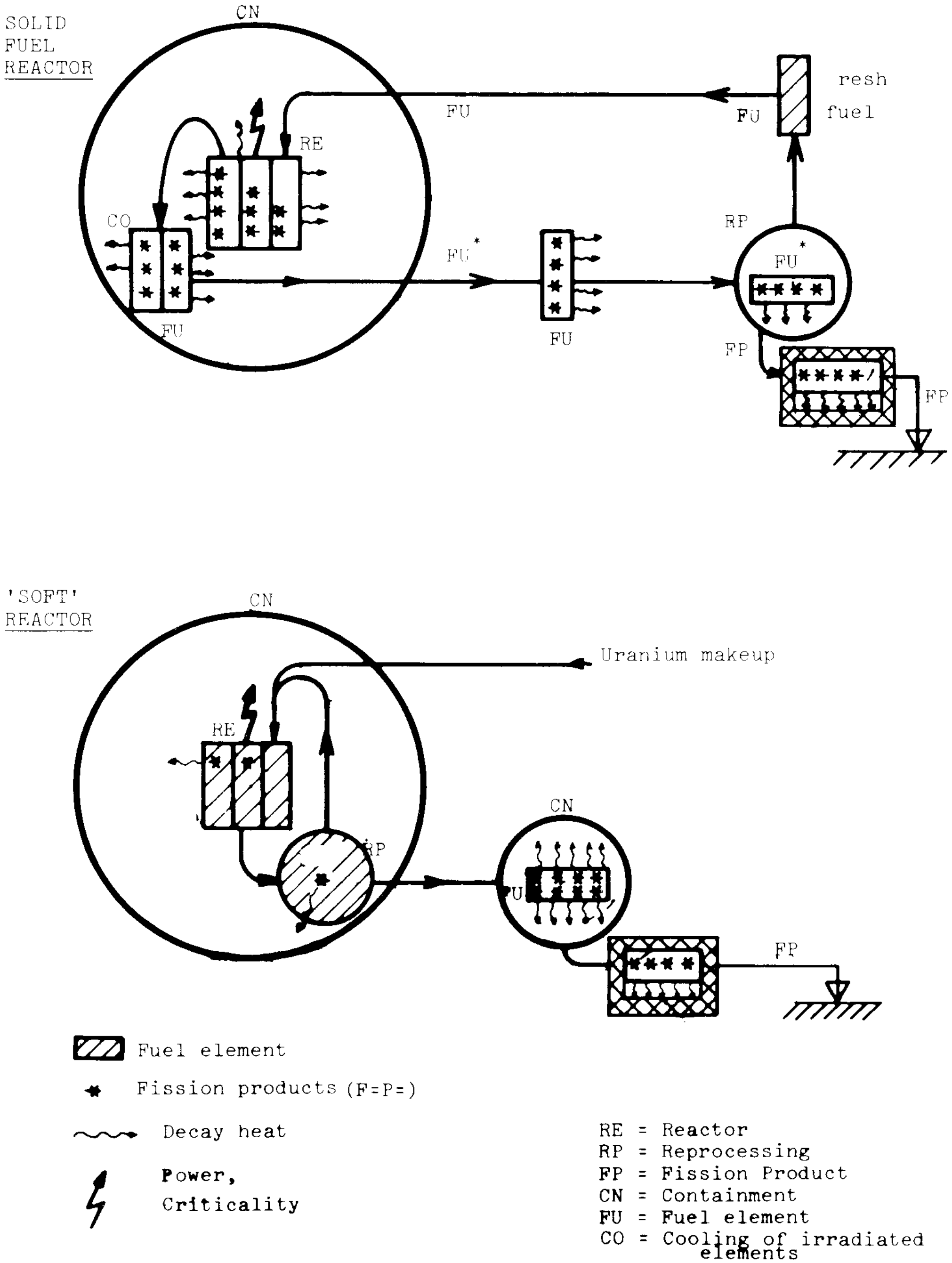
For the accident described above to occur in its entirety it it is necessary that the last barrier, the containment building becomes breached.

Table 2.1 The most dangerous Fission Products according to (WASH 1400)

Release Category	Critical Organ	Nuclides	Relative Dose	Proposed Counter-measure
Radiation from passing cloud from reactor, hours after release	bone marrow	I-132,135,133 I-131 Kr-88,Te-132	~500	continuous extraction of volatile F.P. by means of He-gas purging
	lung	I-132,135,133 I-131,Kr-88 Te-132,Sb-129	~340	
	lower large intestine	I-132,135,133 I-131,Sb-129 Te-132,131m	~240	
	thyroid gland	I-132,135,133 I-131,Kr-88 Te-132,Sb-129	~330	
Long term effect (0-10 yrs) of inhaled radio nuclides 15 /km from reactor	bone marrow	Sr-90,Cs-134 Cs-137 Ce-144,Ru-106	6.7	continuous extraction of non-volatile F.P. in the fuel reprocessing plant
	lung	Ru-106,Ce-144	~126	
	mineral bone	Cs-134,137 Sr-90,Ce-144	~18	
	breast	Cs-134,137 Ru-106,Ce-144	~1.25	

Fig. 3.1

'SOFT'REACTOR VERSUS CONVENTIONAL REACTOR



This can occur in the following ways:

a) steam explosion,
b) leakage of pipes,
c) hydrogen burning,
d) overpressure due to heating,
e) melting of the core systems,
f) loss of fluid giving overpower (for fast reactors).

It seems that in the reactor system proposed here almost all of these failure mechanisms (excluding the leakage of pipes) could be fully eliminated. The safety is thus significantly improved.

3. PRINCIPLES OF THE 'SOFT' REACTOR

3.1 A Schematic View

Of course there is no possible way to build a fission reactor in which the dangerous fission products are not produced at all, or in which they could be fully destroyed (by transformation) in situ. Therefore the only possibility is to shift in time and space:

a) fissionable fuel as a critical source and generator of power - both being controllable,
b) fission products as a source of radioactivity and source of decay heat, both of which cannot be controlled.

Figure 3.1 shows this principle.

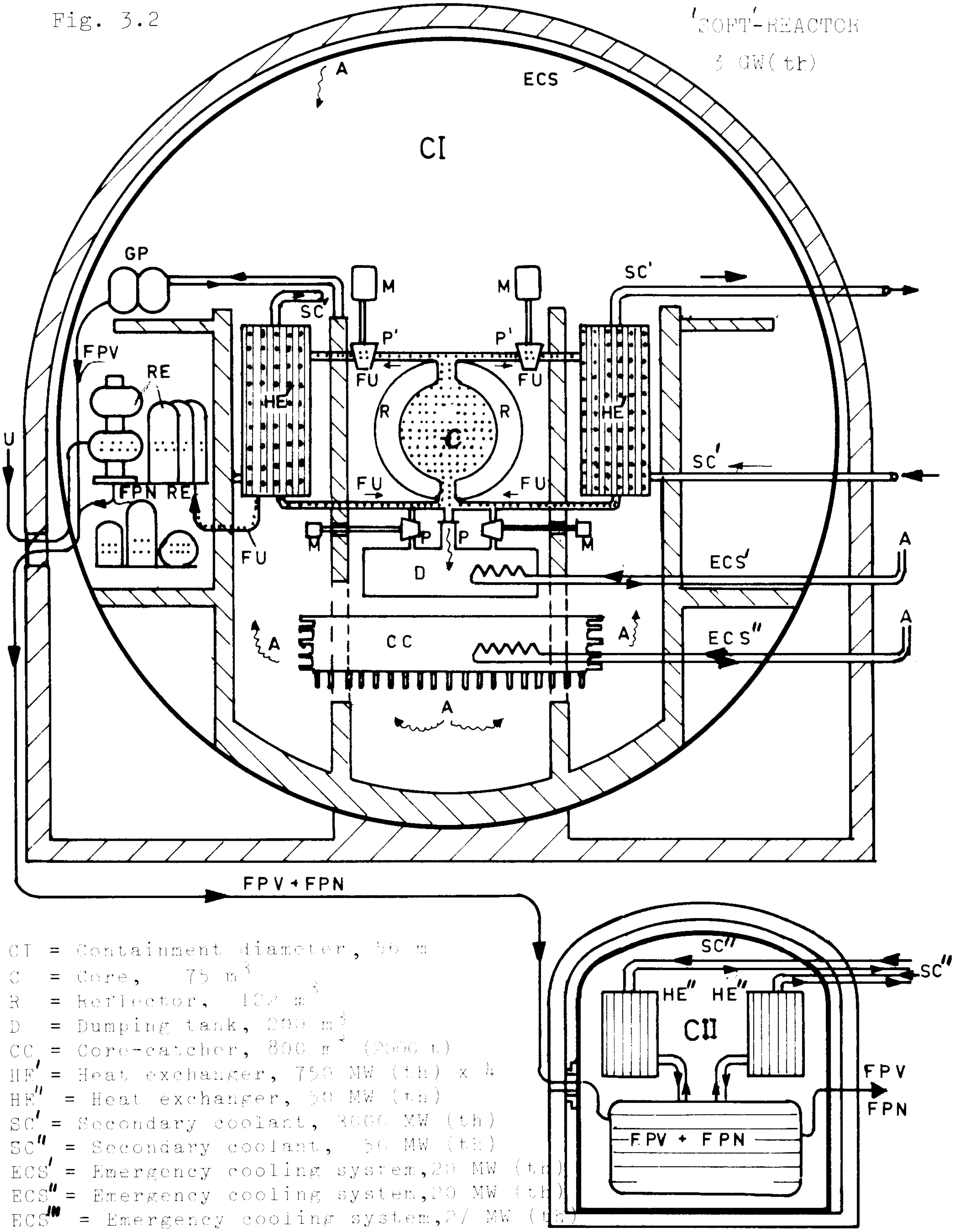
3.2 System Description

For the sake of simplicity, and by no means as an optimised system the reactor characteristics can be summarised as follows: (Table 3.1 and Fig. 3.2)

- fast converter reactor: internal breeding ratio of $\gtrsim 1$; includes a continuously operating reprocessing plant,
- thermal power : 3GW

Fig. 3.2

SCHEME OF
'SOFT'-REACTOR
3 GW(th)



CI = Containment diameter, 56 m
C = Core, 75 m^3
R = Reflector, 122 m^3
D = Dumping tank, 200 m^3
CC = Core-catcher, 800 m^3 (2000 t)
HE' = Heat exchanger, 750 MW (th) x 4
HE'' = Heat exchanger, 50 MW (th)
SC' = Secondary coolant, 3000 MW (th)
SC'' = Secondary coolant, 50 MW (th)
ECS' = Emergency cooling system, 20 MW (th)
ECS'' = Emergency cooling system, 20 MW (th)
ECS''' = Emergency cooling system, 2/ MW (th)
RE = Reprocessing plant
GP = Gas Purging system (Helium)
FPV = Fission products, volatile
FPN = Fission products, nonvolatile
U = Uranium make-up
FU = Fuel

A = Air cooling
M = Motor
P = Pump for fuel
p = Auxilliary pump

CII-Containment, diameter : 15 m)

Table 3.1 A Short Description of the SOFT Reactor

	Unit	Value
Reactor		
Type of reactor	-	fast
Type of fuel	-	molten salt
Type of cooling	-	external cooling
Power - total	GW(th)	3
- electrical gross	GW(e)	1.2
- efficiency	%	40
Fuel		
- composition		1 $PuCl_3 \cdot 8\ UCl_3 \cdot 10\ NaCl$
- density	kg/liter	∿3.3
- total mass	kg	247200
- mass plutonium fissile	w% fuel;kg	5.18w% ;12800
- mass plutonium total	w% fuel;kg	6.48w% ;∿16020
- outlet temperature	°C	650 } ΔT = 180
- temperature, inlet	°C	470
- specific heat of fuel core	GW(th)/K	0.188
- volume flow through core	m^3/s	6.65
- specific power	kW/kg	12.14
- total mass (in core + heat exch)	kg	353 000
- total volume (in core + heat exch)	m^3	107
- breeding ratio	BR	∿1.04
Core		
- geometry	-	spherical
- radius	m	2.616
- volume	m^3	74.99
- specific power	kW/liter	40.00
- dwell time in core	s	∿11.3
- dwell time ex-core	s	4.8
- total circulation time	s	16.1
- neutron flux	$n/cm^2 s$	5×10^{15}
Coolant System		
- primary heat exchanger	number	4
- secondary coolant	-	molten chlorides
- amount of fuel ex-core		
- volume (Fig.6.1 page 48)	m^3	∿32
- mass	kg	106 000
- volume flow in core	m^3/s	6.65
- mass flow in core	kg/s	21 950
- dwelling time in-core	s	11.3
- dwelling time ex-core	s	4.8
- total recirculation time	s	16.1
- amount of coolant		
- specific heat	kJ/liter.K	1.85
- density	kg/liter	1.85
Reprocessing		
- total amount of fuel in system	kg	353 000
- total amount of plutonium	kg	22 874
- total amount of plutonium	moles	∿95 000

- thermal power : 3 GW,
- fuel : molten chlorides : 1 $PuCl_3$ · 8 UCl_3 · 10 NaCl,
- the reactor is cooled externally. This means that part of the molten fuel is pumped out of the core through a heat exchanger,
- the specific power in normal operation is very low 40 kW/liter of fuel in the core,
- fuel is at a rather low temperature of 650°C,
- the internal heat sink is very high: the fuel itself has a heat capacity of approx 80 GJ(th) per 300 K temperature increase.

The negative properties of 'SOFT'

- large plutonium inventory: 22.8 tons, in no way optimized
- low specific power per unit of fissile material,
- a new and unexplored reactor type,
- serious corrosion problems.

3.3 Mass Flows and Containments

This kind of reactor is characterised by the existence of three separate mass flows (see Fig. 3.3):

- the flow of the molten fuel through the external heat exchangers (approx 6.65 m^3/s).
  The ratio of the dwell time outside and in the core is 4.8 s/11.3 s = 0.42
- the flow of molten salt through chemical reprocessing is 0.25 liter/s. The ratio of the dwell times in the reprocessing plant (5 day cycle) and in the core is 2.4 hr/5 x 24 hr. = 0.02

The fission products of concern in the case of an accidental release can be removed by continuous gas purging with helium in the core.

Fig. 3.3

MASS FLOW IN "SOFT"

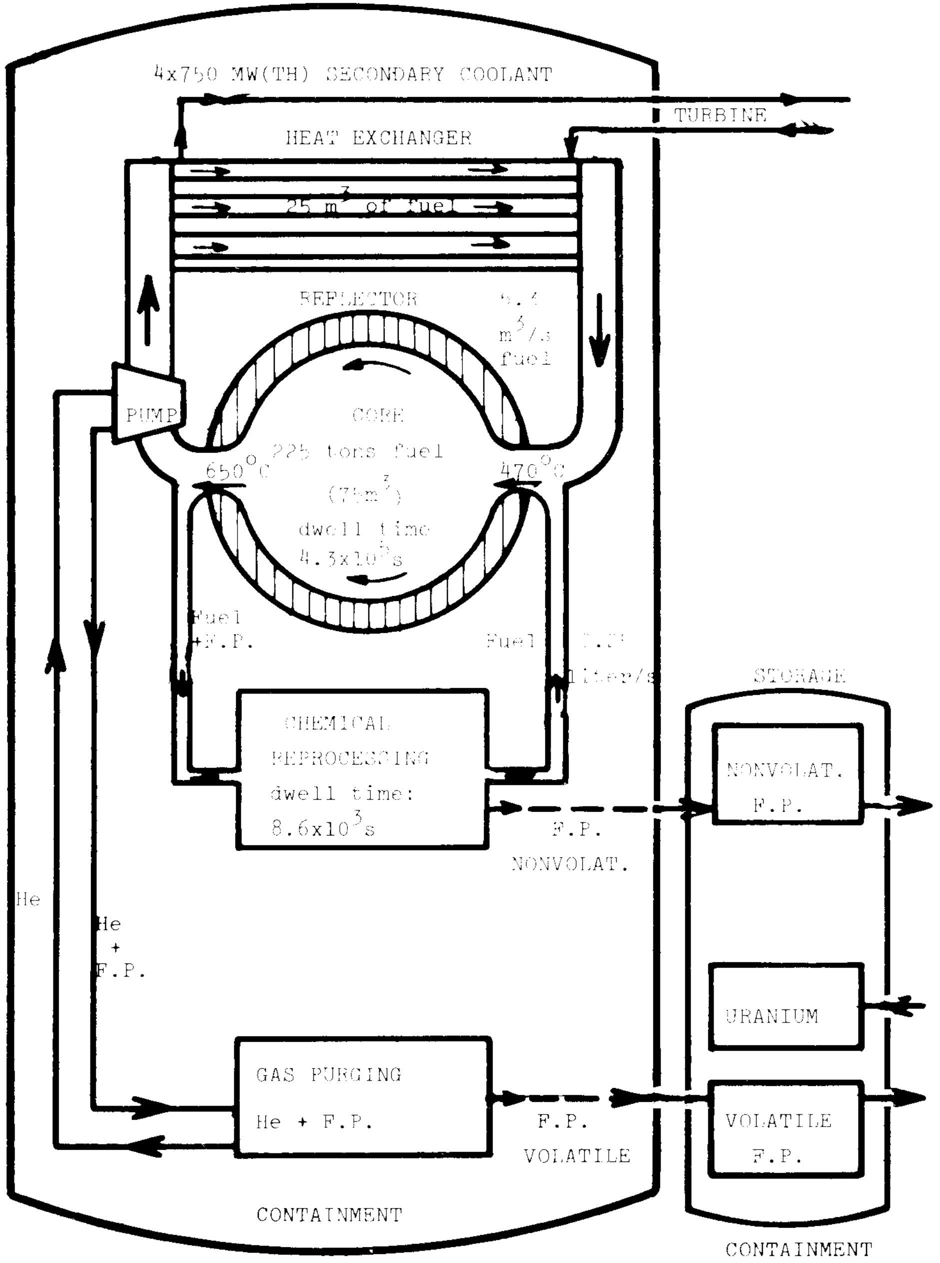


Table 4.1 Neutron Balance of 'SOFT' Reactor

Nuclide	Core			Reflector	Total		
	Product	Fission	Capture	Capture	Product	Fission	Capture
U235	0.046	0.019	0.005	0	0.046	0.019	0.005
Pu239	0.713	0.242	0.052	0	0.713	0.242	0.052
U238	0.140	0.051	0.350	0	0.140	0.051	0.350
Pu240	0.021	0.007	0.009	0	0.021	0.007	0.009
C	0	0	0	0	0	0	0
Na	0	0	0.002	0.001	0	0	0.003
Fe	0	0	0	0.042	0	0	0.042
Pu241	0.065	0.022	0.004	0	0.065	0.022	0.004
Pu242	0.015	0.005	0.005	0	0.015	0.005	0.005
F.P.	0	0	0.001	0	0	0	0.001
Mo	0	0	0	0	0	0	0
B10	0	0	0	0	0	0	0
K	0	0	0	0	0	0	0
Ca	0	0	0	0.000	0	0	0
Cl	0	0	0.105	0.054	0	0	0
Total	1.000	0.346	0.533	0.097	1.000	0.346	0.630

4. NUCLEAR CALCULATIONS

4.1 The Method

The calculations were based on the British 2240 group nuclear data library FGL5. Twenty group cross-sections for each reactor zone were derived from FGL5 using the associated cell code MURALB.

The reactor geometry was represented by a spherical model with one core zone and a reflector zone. Transport calculations were performed in the discrete ordinate (S^4) approximation by the ANISN code to obtain the physical properties of the critical reactor. The effective delayed neutron fraction was derived from basic ENDF-B-IV data using proper flux and adjoint flux weighting and an appropriate reduction of the delayed neutron yields due to precursor losses.

The main results of the physics calculations can be summarised as:

critical radius	2.616 m
average neutron flux	$5 \times 10^{15} n/cm^2 s$
temperature criticality coefficient	0.018% /K

(see also Table 4.1 and Table 4.2)

Effect of doppler contribution

- not calculated but according to ref. 9 is very small

Table 4.2 Calculated One Group Cross Sections

	Nuclide	Concentration $(10^{24} cm^{-3})$	$\nu\sigma^f$ (barn)	σ^f (barn)
CORE	U235	0.31 - 4	4.109	2.082
	Pu239	4.04 - 4	4.919	2.032
	U238	42.75 - 4	0.916 - 1	0.262
	Pu240	0.54 - 4	1.066	0.801
	C	1.0 - 18	0	0.470 - 4
	Na	53.83 - 4	0	0.885 - 3
	Fe	1.0 - 18	0	0.996 - 2
	Pu241	0.27 - 4	6.763	2.736
	Pu242	0.54 - 4	0.791	0.524
	F.P.	0.11 - 4	0	0.191
	Mo	1.0 - 18	0	0.115
	B10	1.0 - 18	0	2.146
	K	1.0 - 18	0	0.172 - 1
	Ca	1.0 - 18	0	0.237 - 3
	Cl	199.39 - 4	0	0.147 - 1
REFLECTOR	U235	1.0 - 18	10.403	6.154
	Pu239	1.0 - 18	11.597	6.595
	U238	1.0 - 18	0.123 - 1	2.541
	Pu240	1.0 - 18	0.576	4.076
	C	1.0 - 18	0	0.120 - 4
	Na	33.65 - 4	0	0.287 - 2
	Fe	420.60 - 4	0	0.136 - 1
	Pu241	1.0 - 18	24.676	10.920
	Pu242	1.0 - 18	0.261	2.384
	FP	1.0 - 18	0	1.009
	Mo	1.0 - 18	0	0.644
	B10	1.0 - 18	0	9.301
	K	1.0 - 18	0	0.245 - 1
	Ca	33.65 - 4	0	0.865 - 3
	Cl	100.9 - 4	0	0.736 - 1

4.2 Nuclear Properties of SOFT

The loss of neutrons due to the use of natural chlorine are 10% which is significantly high but far from being prohibitive. (Ref. 8, 10, 16, 20, 21).

5. CONTINUOUS EXTRACTION OF FISSION PRODUCTS

5.1 General Scheme

The general idea of this reactor type is the differentiation of the chemical properties of the fission products in the molten chloride medium. (Ref. 10, 16).
The rough scheme is given in Fig. 5.1.

5.2 Chemical State of Fission Products in Molten Chlorides

It is important to know the chemical state of the fission products. The fission reaction of $PuCl_3$ can be written as follows (El = fission product element) (index 0 for elementary state)

$$1\ PuCl_3 \xrightarrow{\text{fission}} 3/2\ El\cdot Cl_2 + 1/2\ El_0$$

Cumulated Cl-demand is calculated from individual fission product yields (Y) for fast fission of ^{239}Pu irradiated for 10 days and product stoichiometry starting with the most stable chloride product being formed. Where necessary (i.e. where one fission product will form several chlorides of comparable stability) a weighted mean was calculated based on the Gibbs free energies at 1000 K. (see also Ref. 6, 17, 22)

Table 5.1 fives the cumulated chlorine demand of the fission products forming stable chlorides at 1000 K(100% is set equal to the number of fission events - total Cl supply therefore equals 300%). Because of the apparent lack of free chlorine

Fig. 5.1

SIMPLIFIED FLOW OF FISSION PRODUCTS

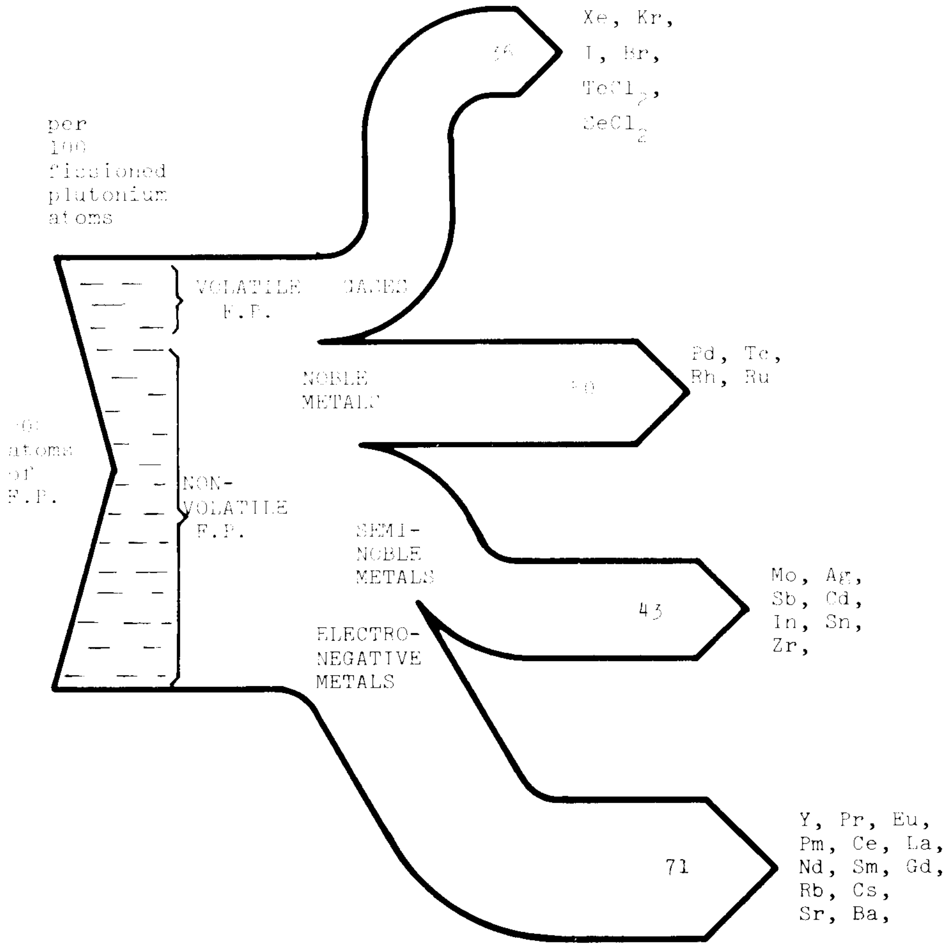


Table 5.1

Cumulated Cl-Demand of Fission Products forming stable chlorides at 1000 K

Compound	Yield (%)	Chlorine demand (%)	Cumulative chlorine demand (%)
$BaCl_2$	9.502	19.004	19.00
$RbCl$	1.050	1.050	20.05
$CsCl$	13.355	13.355	33.41
$SrCl_2$	5.487	10.974	44.38
$SmCl_2$	3.737	7.474	51.86
$LaCl_3$	5.794	17.382	69.24
$CeCl_3$	13.986	41.958	111.20
$PrCl_3$	4.278	12.834	124.03
$NdCl_3$	11.870	35.61	159.64
YCl_3	3.028	9.084	168.73
$ZrCl_{2.6}$*	21.520	63.05	231.78
$InCl_{1.3}$*	0.060	0.078	231.85
$CdCl_2$	0.661	1.322	233.18
$SnCl_2$	0.324	0.648	233.82
$SbCl_3$	0.674	2.032	235.86
$AgCl$	1.880	1.880	237.74
$SeCl$	0.088	0.088	237.75
$TeCl_2$	7.654	15.308	253.06
$NbCl_5$	0.289	1.445	254.51
$MoCl_{2.9}$*	18.16	34	288.51
$PdCl_2$	12.66	8.8	297.31
$TcCl_2$	4.01	2.7	300.00

most stable

ΔG^{1000} → 0

least stable

* weighted mean

Δ_G^{1000} = Gibbs free energy at 1000 K only 65% of the molybdenum and only 35% of the palladium and technetium being produced in the fission process will form chlorides. These cations cannot form stable iodides at 1000 K (their respective ΔG_f^o being in the range from + 48.5 $\frac{kJ}{mol\ Cl}$ to 67.4 $\frac{kJ}{mol\ Cl}$ (Fig. 5.2) and will probably be present in metallic form.

5.3 Fission Product Volatility and Gaseous Extraction

Table 5.2 gives the melting and boiling points of the fission product chlorides and the fuel components. (Ref. 22)

The chemical species in the system discussed here can be divided into 3 groups :

- those that have melting points of 1000 K are likely to solidify under normal operating conditions (e.g. semi-noble metals),
- those which have a boiling point 1000 K (or 1500 K) will evaporate under normal operating conditions (in case of accidental use in temperature) and will have to be removed from the reactor altogether with other gaseous products (i.e. Xe, Kr, I_2 and He as an inert carrier gas). This will necessitate the installation of appropriate equipment to extract these compounds from the purge gas before it is released to the environment,
- all other chemical species.

The volatile fission products Xe, Kr, I, Te and Sb are of primary concern because they would be almost completely released in a core accident. The ten radionuclides that contribute the most (92% of the total) of the external dose to the bone marrow from a passing cloud (see Fig. 5.3) were

Fig. 5.2
FREE ENERGY OF FORMATION OF
IODIDES AND CHLORIDES

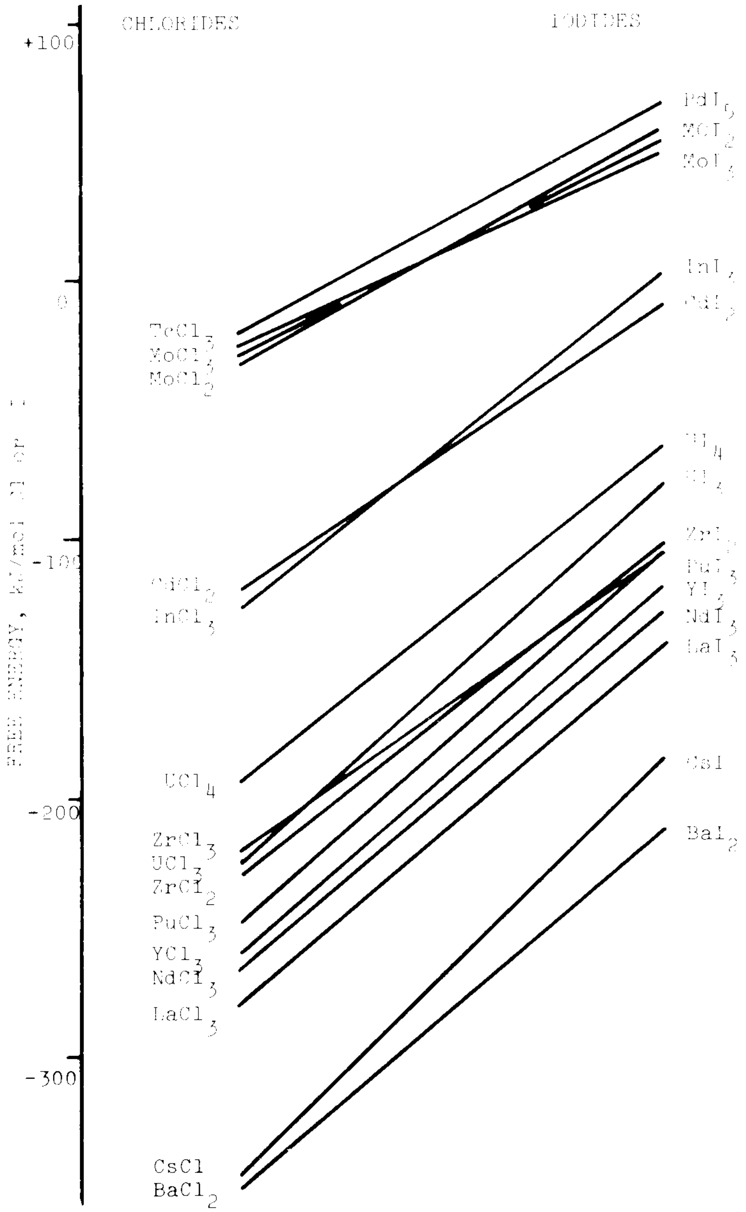


Fig. 5.3
RADIATION DOSE FROM A PASSING
CLOUD BY CONTINUOUS EXTRACTION
OF VOLATILE RADIONUCLIDES

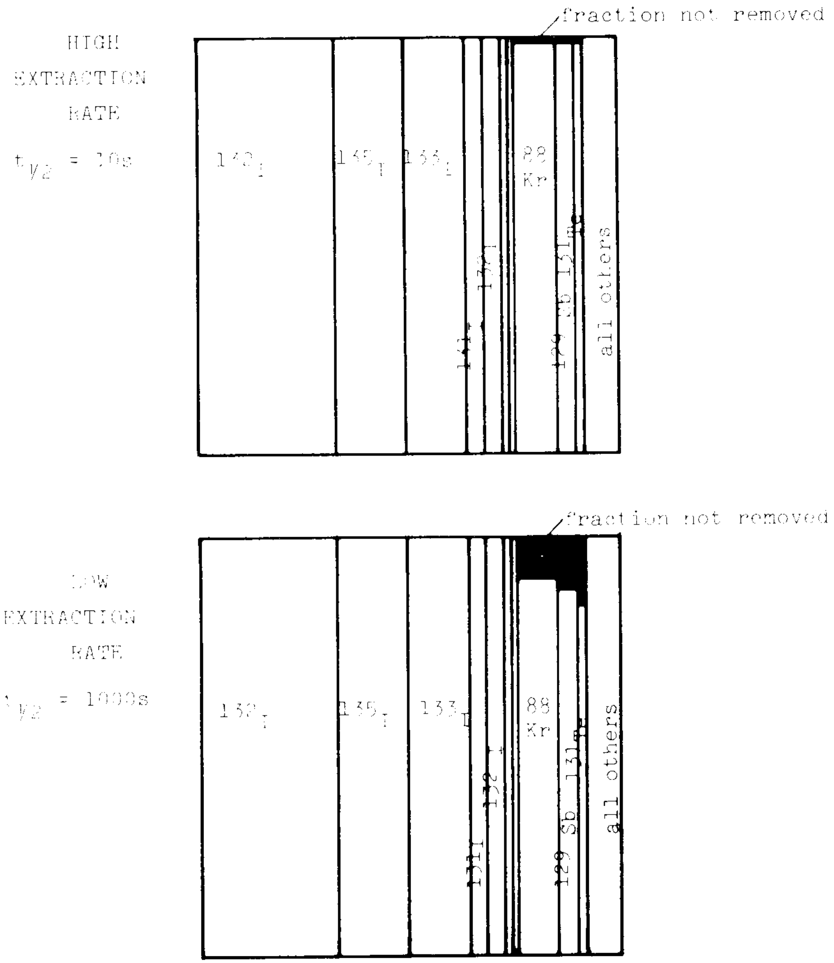


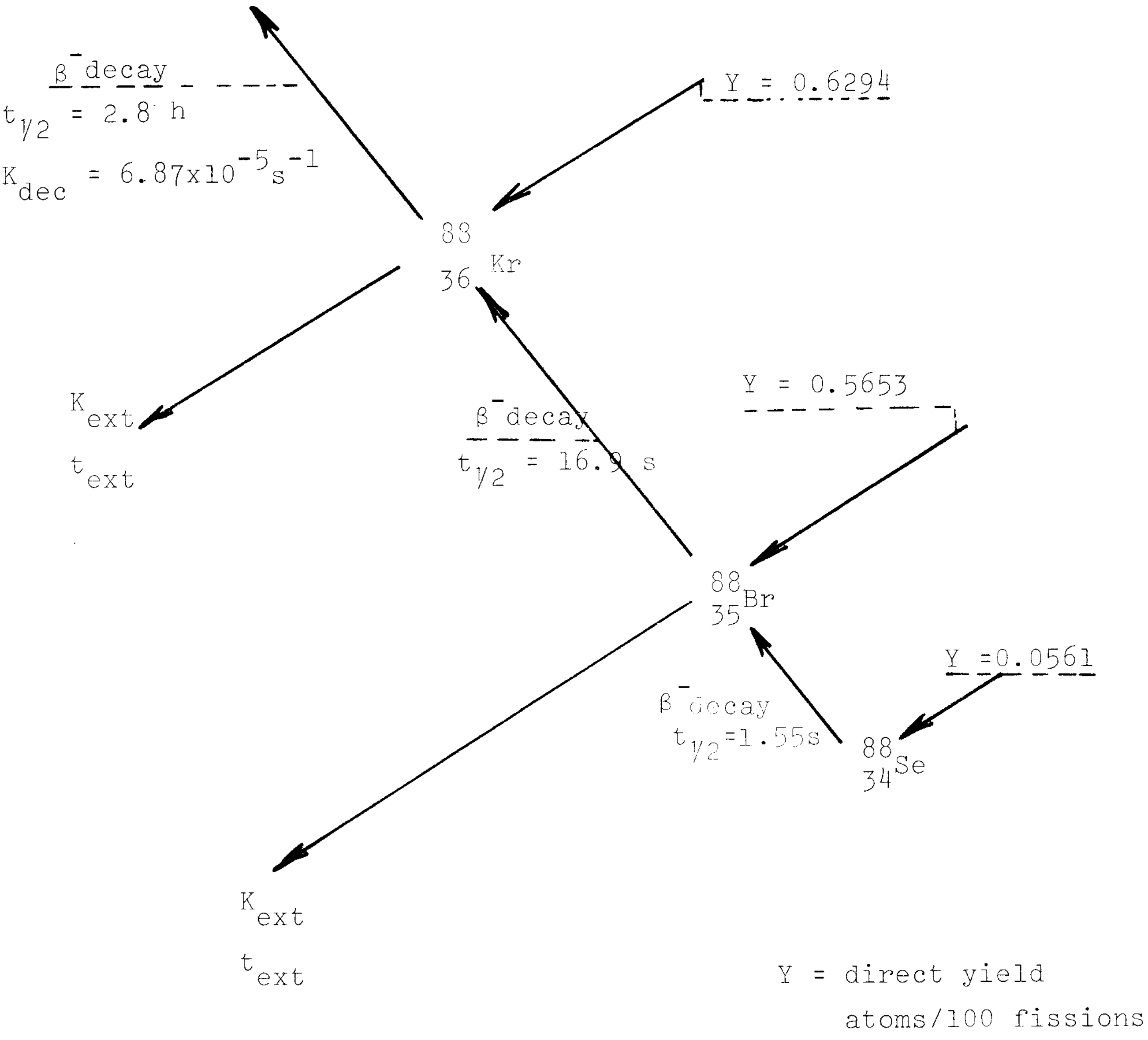
Table 5.2

Melting- and Boiling-Point of Fission-Product-Chlorides at Standard Pressure.

Compound	Melting point (K)	Boiling point (K)	Reference
SeCl		403	3
$SeCl_4$	469	Diss	3
RbCl	990	1654	3
$SrCl_2$	1145	2300	3
YCl_3	973	1725	3
$ZrCl_2$	1000	1660	3
$ZrCl_3$	900	1480	3
$ZrCl_4$		604	3
$NbCl_5$	478 - 480	489.6	6
$MoCl_2$		Diss.	3
$MoCl_5$	467	541	6
$MoCl_4$	590		6
$MoCl_3$		Diss.	3
$MoCl_6$	580	630	3
$TcCl_3$	1220	1475	3
$RuCl_3$	900	Diss.	3
RhCl	1000	1238 Diss.	3
$RhCl_2$	1050	1231 Diss.	3
$PdCl_2$	951	1300	3
AgCl	728	1837	3
$CdCl_2$	841	1248	3
InCl	498	881	3
$InCl_2$	508	758	3
$InCl_3$		771	3
$SnCl_2$	520	925	3
$SnCl_4$		386	3
$SbCl_3$	346.4	492	3
$SbCl_5$		445	3
$TeCl_2$	448	595	3
$TeCl_4$	497.3	665	3
CsCl	918	1573	3
$BaCl_2$	1198(α) 1235 (β)	2100	3
$LaCl_3$	1125	2020	3
$CeCl_3$	1085	2000	3
$PrCl_3$	1049	1980	3
$NdCl_3$	1033	1960	3
$SmCl_2$	1013	1700	
$SmCl_3$	951	1830	3
UCl_3	1108	2000	3
UCl_4	863	1060	3
UCl_5	560	690	3
UCl_6	452	550	3
$PuCl_3$	1033	2040	3
$PuCl_4$	730	1060	3
NaCl	1074	1686	3

chosen to study the effectiveness of gaseous extraction. The extraction rate will be characterised by a rate constant K_{ext} which will depend on the physical and chemical properties of the volatile, the salt melt and design parameters such as gas throughput. For simplicities sake the same rate constant is used in the calculations for all the radionuclides that are studied.

As an example the following simplified scheme is used to calculate the ^{88}Kr and ^{129}Sb inventory:



The halflives of ${}^{88}_{35}\mathrm{Br}$ and ${}^{88}_{34}\mathrm{Se}$ are so short that they will decay to ${}^{88}_{36}\mathrm{Kr}$ before they can be removed from the reactor. The rate of production of ${}^{88}_{36}\mathrm{Kr}$ is therefore equal to the sum of the direct yields of ${}^{88}_{36}\mathrm{Kr}$ ${}^{88}_{35}\mathrm{Br}$ and ${}^{88}_{34}\mathrm{Se}$. These direct yields were calculated as the product of the values given in ref. 4 and represent the number of nucleii formed by fast fission of 100 atoms of ${}^{239}\mathrm{Pu}$ directly before any radioactive decay has taken place.

Similarly the rate at which ${}^{88}_{36}\mathrm{Kr}$ is disappearing is characterised by the sum of the respective rate constants for radioactive β^-decay and extraction, where the half-life is given by $t_{1/2} = \frac{\ln 2}{K}$

Thus the change in ${}^{88}_{36}\mathrm{Kr}$ inventory $({}^{88}\mathrm{Kr})$ is given by :

$$\frac{d({}^{88}\mathrm{Kr})}{dt} = V \cdot F \cdot \left[Y + D - K_{dec}\,({}^{88}\mathrm{Kr}) - K_{ext}\,({}^{88}\mathrm{Kr}) \right]$$

where V is the reactor volume (m^3), F the fission rate (fission/s), Y the direct yield (atoms/fission), D the decay rate of its precursors and $({}^{88}\mathrm{Kr})$ is the concentration of ${}^{88}\mathrm{Kr}$ exposed, in units such as (10^{20}atom/cm^3) or (mol/m^3).

During steady state operation the inventory will remain constant i.e. as many radionuclides are removed from the reactor as are produced in any given time. The equation above can therefore be solved if we assume a value for K_{ext} the only parameter not given by nature.

Arbitrarily setting the product:

(reactor volume) x (fission rate) = 1

i.e. putting the above equation in a dimensionless form we get

$t_{1/2}^{ext}$ (s)	K_{ext} (s^{-1})	(^{88}Kr)
10	6.93×10^{-2}	1.81×10^{-1}
1000	6.93×10^{-4}	1.65×10^{-1}
100000	6.93×10^{-5}	9.03×10^{1}
∞	0	1.81×10^{2}

Similar calculations were made for other extractable nuclides.

The extraction rate cannot be very high not only because of the physical and chemical constraints and the prohibitive amount of helium bubbles in the fuel (effecting the criticality due to changes in total and local bubble volumes) but also because of the problem of the precursors of the delayed neutrons.

5.4 Problem of Delayed Neutron Emitters

A significant proportion of the delayed neutron precursors are the volatile fission products. Delayed neutron emitters will be removed along with other volatile fission products by means of gas purging. The nuclides of interest and their respective half lives are approximately as follows :(see Ref.9)

Nuclide	Neutrons/fission x 100	Half life (s)
^{87}Br	0.024	54.5
^{137}I	0.176	24.4
^{88}Br	0.136	16.3
^{138}I		6.3
^{89}Br	0.207	4.4
^{139}I		2.0
^{90}Br	0.087	1.6
Total	0.630	

^{138}I and ^{139}I have no precursors. ^{137}I and its short-lived precursor ^{137}Te ($t_{1/2}$ = 3.5 s)

The effect of gaseous extraction upon delayed neutron emitters inventory can be calculated roughly by the same method used in section 5.4 to calculate the degree of extraction of the other volatile F.P. The results of these calculations are presented in Fig. 5.4 and Fig. 5.5. (Ref.17)

Delayed neutrons will also be lost in the heat exchanger as the salt melt is pumped through this loop. The salt melt will remain an average of 11.3 s in the core and 4.8 s in the external heat exchanger circuit. During these 5 s the number of radionuclides is reduced by a factor of 2 exp $(-5\,s/t_{1/2})$ thus total loss due to β^- decay in the heat exchanger circuit is:

$$Y \cdot (1 - X_{ext}) \cdot \frac{4.8}{16.1} \cdot \left(1 - 2 \exp\left(\frac{-4.8\ s}{t_{1/2}}\right)\right)$$

With a time constant of extraction t_{ext} arbitrarily chosen as 10^2s the following results can be obtained :

Radio-nuclide	Relative yield (normalised to Σ = 1)	Loss due to β-decay in external heat exchanger	Loss due to extraction	Total loss to yield ratio	"Weighted" losses
^{87}Br	0.129	0.313	0.187	0.500	0.965
^{137}I	0.499	0.289	0.076	0.365	0.182
^{88}Br	0.120	0.269	0.046	0.315	0.038
^{138}I	0.096	0.192	0.018	0.210	0.020
^{89}Br	0.058	0.152	0.008	0.160	0.009
^{139}I	0.080	0.059	0.001	0.060	0.005
^{90}Br	0.028	0.038	0.000	0.038	0.001
Total	1.000	0.250	0.070		0.320

These losses of 0.32 are significant but not prohibitive.

REMOVAL OF DELAYED-NEUTRON-EMITTERS Fig. 5.4

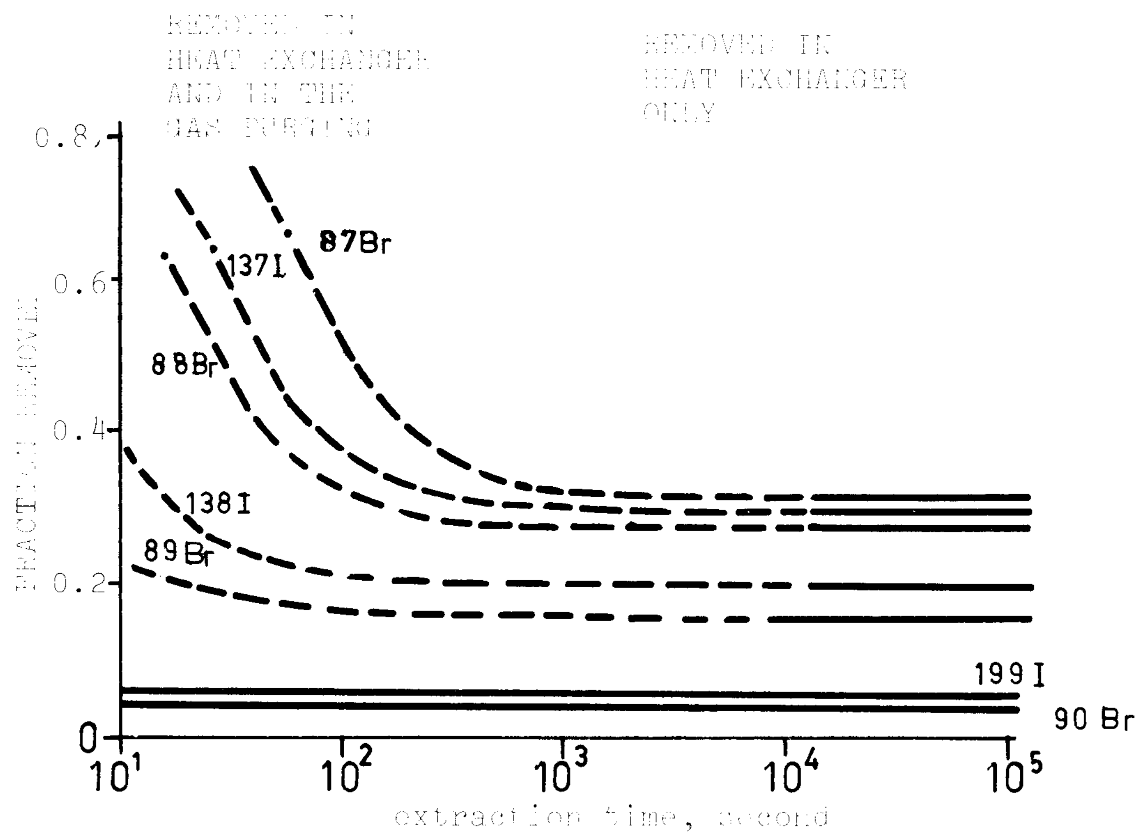
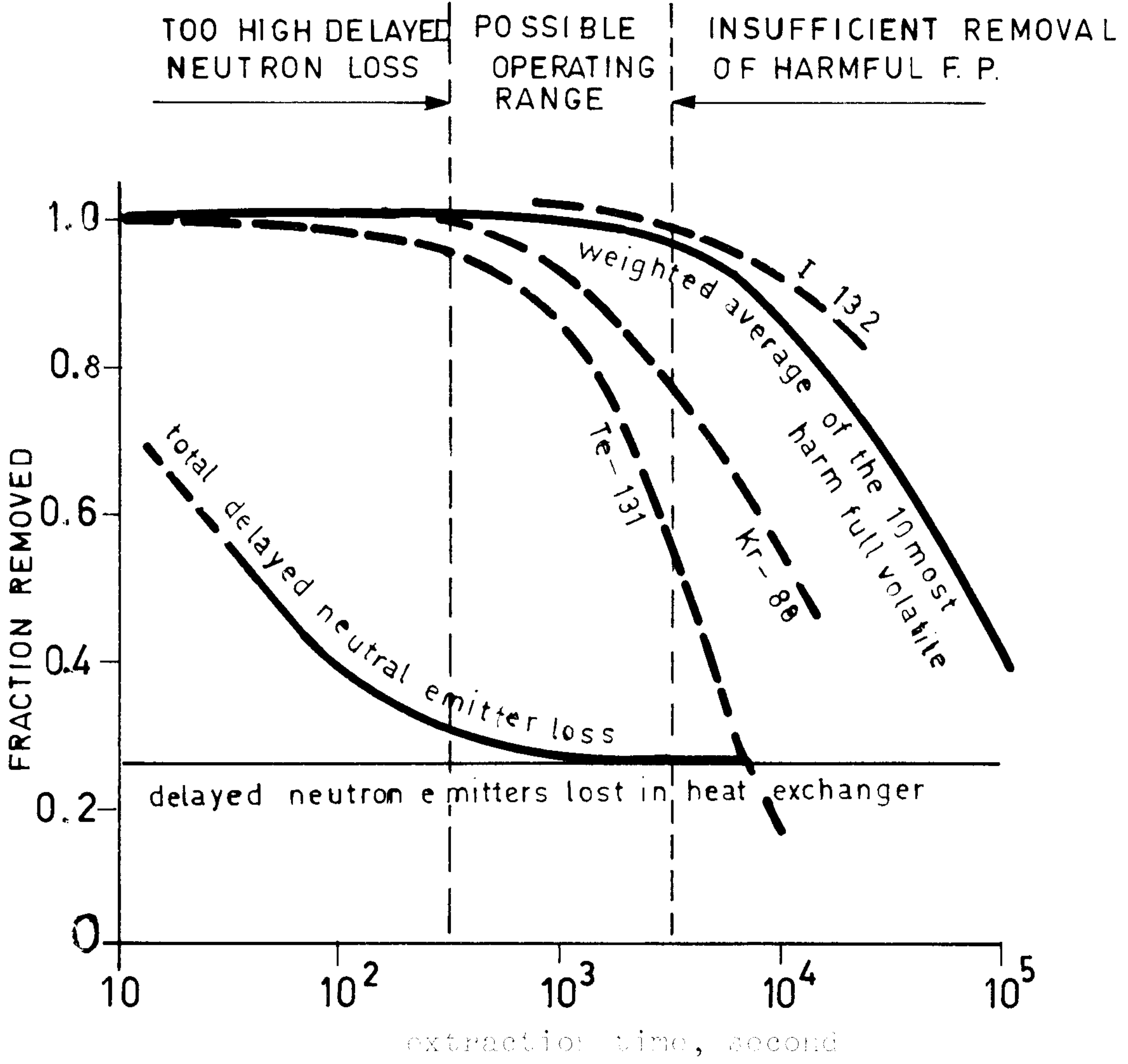


Fig. 5.5



5.5 Gas Extraction Rate

In a 3 GW(th) reactor 9.3×10^{19} fission/s occur producing 3.09×10^{-4} mol/s of fission products, 1×10^{-4} mol/s of which are volatile. The total volume of helium bubbles in the core should not exceed $0.1\,m^3$ (less than 0.2% of the total core volume) that is 5 moles of helium in order to minimise the rise in criticality due to loss of helium throughput.

If we assume the extraction process is limited only by diffusion (i.e. the volatiles are completely immiscible in the molten salt which is likely to be the case for Kr and Xe but hardly for I_2 $TeCl_3$ and $SbCl_3$) the extraction rate can be approximated from film theory. The model assumes that the gas in the bubble undergoes toroidal circulation.

As the gas circulates it encounters fresh liquid at the top of the bubble. As the bubble rises the liquid moves downward in relation to the bubble and leaves when it reaches the bottom of the bubble. The liquid near the interface is usually in laminar flow and the contact time is certainly short, thus the liquid behaves much like the liquid at the surface of a falling film. Therefore to a first approximation the average rate of mass transfer across the interface $(N_A)_{avg}$ is: (Ref.2)

$$(N_A)_{avg} \doteq \frac{4\theta_{AB}}{\pi \cdot t_{cont}} \bullet C_{AO}$$

in which C_{Ao} is the concentration of volatile A in the salt melt, θ_{AB} its diffusion coefficient in the melt and t_{cont} the contact time given approximately by $t_{cont} = \frac{D}{V_t}$ where D is the bubble diameter and V_t the terminal velocity.

During the normal operation of the reactor the volatiles produced in the fission process are removed as quickly as they are produced.

Therefore:

$$F = S \cdot N \cdot (N_A)_{avg}$$

Where F is the fission rate, S the surface of 1 bubble and N the number of bubbles in the core.

A uniform bubble diameter of 3 mm was assumed and θ_{AB} calculated from the Stokes-Einstein equation. For a typical volatile molecule with a molecular weight of ∿30 the value of θ_{AB} is $\sim 2x10^{-9}$ m^2/s. C_{Ho} thus equals $7x10^{-6}$ mol/m^3 equivalent to the production for 7 seconds. (Ref.2, 23)

This investigation shows that:

- gaseous extraction of volatile fission products would be an effective means of reducing the risk to the environment due to an accidental release of part of the reactor inventory in a MSFBR,
- the extraction rates which would be necessary to achieve this should be attainable with a reasonably small throughput that should minimise any criticality change and allow a reasonable design of a gas purification system,
- delayed neutron emitters are removed less than the volatiles that cause concern in the case of a core accident because they have few (or no) volatile precursors and are short-lived compared to the time constant of extraction.

5.6 Reprocessing of Non Volatile Fission Products.

The fission products experience the conditions in the molten chloride core of the reactor and are at ∿ 650°C and in the non-volatile form, e.g:

- dispersions of metallic particles (colloids?) Rh, Ru,
- solutions of molten chlorides: $CsCl$, $SrCl_2$, $RECl_{2 \div 3}$.

They must be extracted from the irradiated fuel by means of chemical reprocessing, if possible continuously. A rough list of the different fission products which may be formed in the molten chloride reaction is given in Table 5.3.

Table 5.3 Extraction of Fission Products in 'SOFT'

Element/ Isotope	Production of element (yield per 100 fissions)	Principle chemical species at 950 K	ΔG_f^o of chloride at 1500 ($\frac{kJ}{mol\ Cl}$)	T_b at 1 bar (K)	Decon-tamination factor *
Sr 90	5.487	$SrCl_2$(l)	-320	2300	100 R
Rb 86	1.050	RbCl (l)	-275	1654	3.3 R
Cs 134	13.355	CsCl (l)	-275	1573	125 R
Cs 137					
La 140	5.794	$LaCl_3$(l)	-255	2020	1.2 R
Ce 144	13.986	$CeCl_3$(l)	-245	2000	20 R
Pr 143	4.287	$PrCl_3$(l)	-245	1980	2.6 R
Y 90	3.028	YCl_3(l)	-240	1725	1.3 R
Nd 147	11.870	$NdCl_3$(l)	-235	1960	2.4 R
Sb 129	0.674	$SbCl_3$(g)	- 80	492	8.4 R
Te 131m	7.654	$TeCl_2$(g)	- 62	595	6.1 G
Te 132					250
Tc 99m	4.010	$TcCl_2$(l)	-12	1474	100 R
Ru 105	31.445	Ru_{met}(s)	0	4170	1.0 R
Rh 105	1.736	Rh_{met}(s)	0	4000	1.2 R
Xe 133	21.234	Xe (g)	0	166	500 G
Xe 135					100
I 131	6.177	I_2 (g)	0	457.5	500 G
I 132					100 G
I 133					500 G
I 134					100
I 135					100 G
Kr 88	0.942	Kr (g)	0	121	11 G

Remarks

(l) liquid (g) gaseous

* The "decontamination factor" is defined as the ratio of inventories without and with extraction.

The letters R and G refer to chemical reprocessing and gaseous extraction respectively

ΔG_f^o = free energy of formation at 1500 K in kJ/mol Cl

T_b = boiling point at 1 bar

5.7 Some Technological Problems

Apart from the fission products and corrosion products an important problem in this molten chlorides reactor is the removal of sulphur. In a flux of fast neutrons the following reaction takes place: (Ref. 8, 18, 20)

$$^{35}_{17}\mathrm{Cl}\ (n,p)\ ^{35}_{16}\mathrm{S}\quad (t_{1/2} = 88\ \mathrm{d})$$

This sulphur is obviously undesirable and makes the corrosion problems worse. The continuous chemical reprocessing can be used to remove the sulphur. Nevertheless a considerable problem remains regarding the resulting corrosion.

5.8 Possible Reprocessing Technologies

It is possible to separate the fission products from the molten fuel in various ways:

- extraction from molten chlorides by imiscible liquid metals. This is a well known technology developed especially for the molten salt breeder (Ref. 13)
- electrochemical dissolution of metallic uranium. The continuous feed of uranium (depleted or natural) in amounts of some gram/sec permits the removal of fission products which have a free energy of formation of chlorides of ∿1000K which is smaller than that of UCl_3. This applies to a good proportion of the semi-noble F.P. (see Fig.5.1)
- the most attractive method is electrolysis in the molten chloride system. The following very simplified calculation explains this: The total amount of plutonium in the fuel is 96 000 moles. (see Table 3.1). The total amount of chlorides is 1.82×10^6 moles ($PuCl_3$, UCl_3, NaCl). For a five day reprocessing cycle (4.32×10^5 s) the flow to the reprocessing plant is 4.21 mol/s.
  Taking into account the electrolysis is at a voltage of 3 volts (see Fig. 5.5) and the efficiency is 0.5 then the power needed for the electrolysis is:

Fig. 5.5

ELECTRICAL POTENTIAL IN MOLTEN CHLORIDES

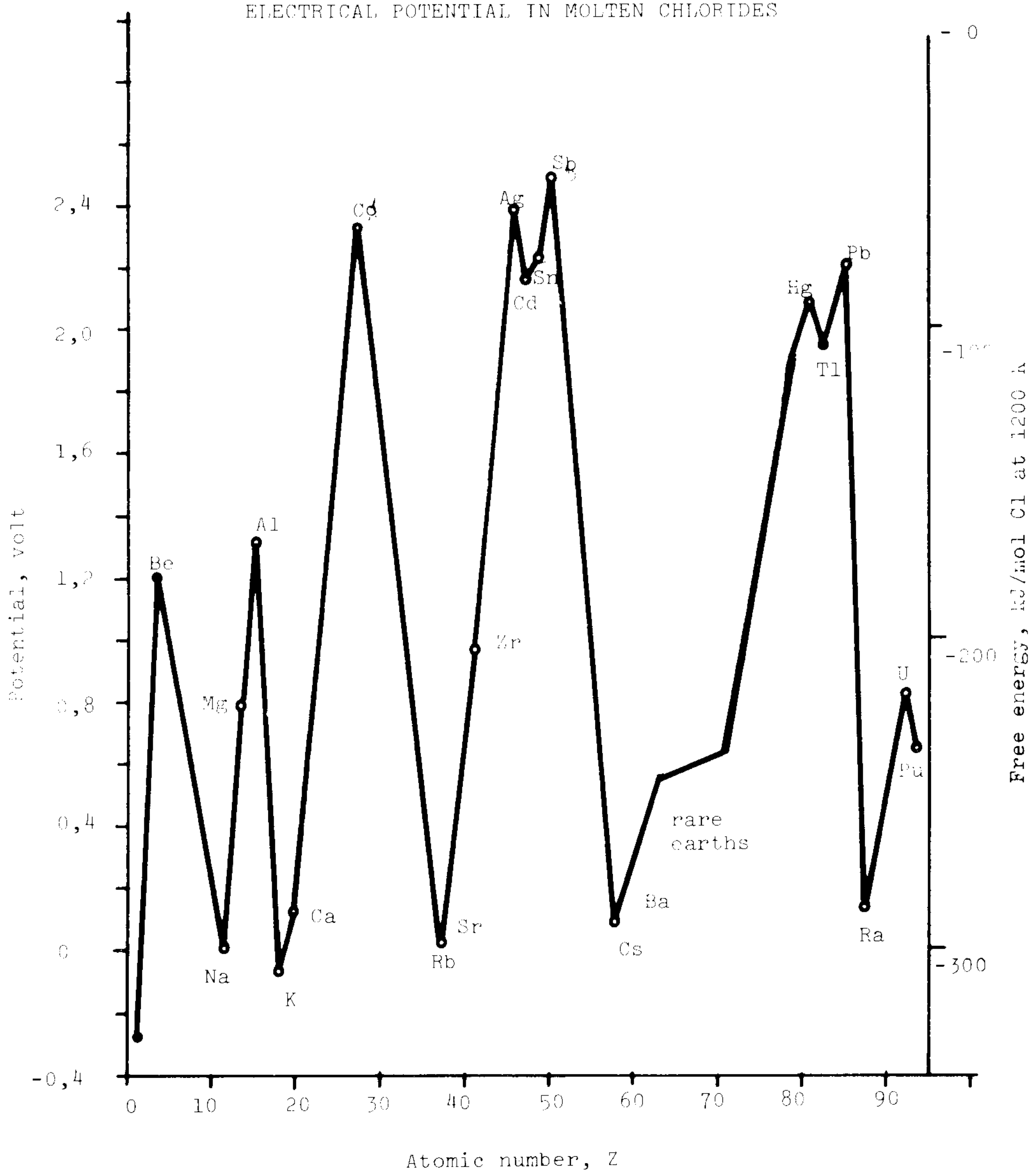
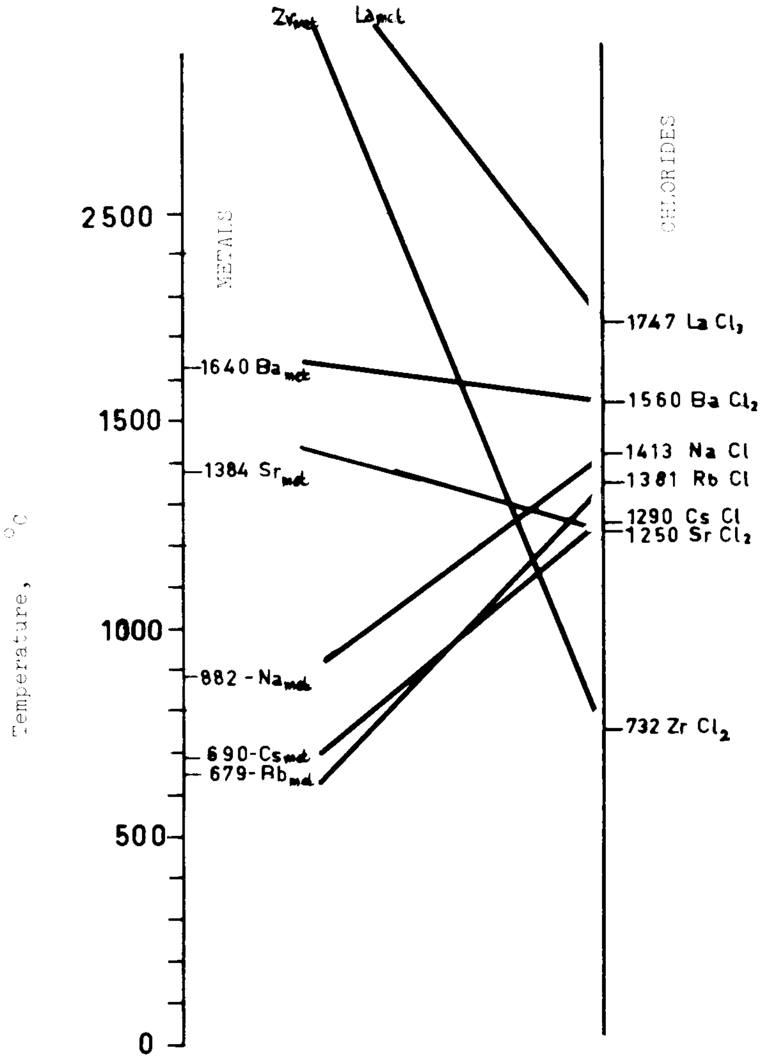


Fig. 5.6

BOILING POINT OF METALS AND CHLORIDES



$$4.21 \text{ mol/s} \times 3\text{ V} \times 96\,500 \text{ C/mol} \times \frac{1}{0.5} = 2.44 \text{ MW(e)}$$

The amount of electrical energy needed for the electrolysis of the total amount of fuel, related to the total power production is only:

$$\frac{2.44 \text{ MW(e)}}{3000 \text{ MW(th)} \times (\eta_e \approx 0.5)} = 0.16 \text{ percent}$$

Continuous electrolysis seems to be feasable both from the economic and technical point of view.

5.9 The Problem of External Storage of the Fission Products

For the reactor type proposed a special problem has to be solved: that of the interim storage of the fission products from the time of extraction from the fuel to the time of transportation away from the reactor building approximately 30 days later.

The criteria giving the form of chemical and physical form best suited for interim storage is shown in Table 5.4.

6. HEAT REMOVAL

6.1 Primary Circuit: Core and Heat Exchangers

Molten fuel reactors have been proposed having differing methods of cooling the primary fuel : a) internal cooling in which the coolant is pumped into the core in for example tubes, b) external cooling systems in which the fuel is pumped out of the core through heat exchangers.

The reactor proposed here has the following characteristics:

- the heat removal is based on the pumping of the molten fuel out of the core. The dwell time in the core is ∿ 11.3 s, and outside the core ∿ 4.8 s.

- the outlet temperature of the fuel is 650°C the inlet temperature 470°C. The temperature difference of 180°C with a volume flow rate of 6.65 m^3/s permits the removal of 3000 MW(th) (see Fig.6.1)
- the secondary coolant is also a molten salt containing chemically inert chlorides of Na, Ca, Ba, Mg. No pressure is needed and no hydrogen containing a volatile compound are present.

6.2 Secondary Circuit

The following are the requirements for a cooling medium for this reactor:

- generating at the proposed 600°C, pressureless and stable for long periods of irradiation.
- non reactive to molten fuel (no exothermic or endothermic chemical reactions, no precipitation or gas evolution.

Some media which fulfill the above requirements are:

Low melting chlorides

mol per cent					T_{melt} (°C)
NaCl	KCl	$MgCl_2$	$CaCl_2$	$BaCl_2$	
20	20	60	-	-	396
38.5	-	-	47	14.5	450
-	20	60	-	20	440

6.3 The Tertiary Circuit

A rather unconventional choice has been made for the medium for and form of the tertiary circuit.

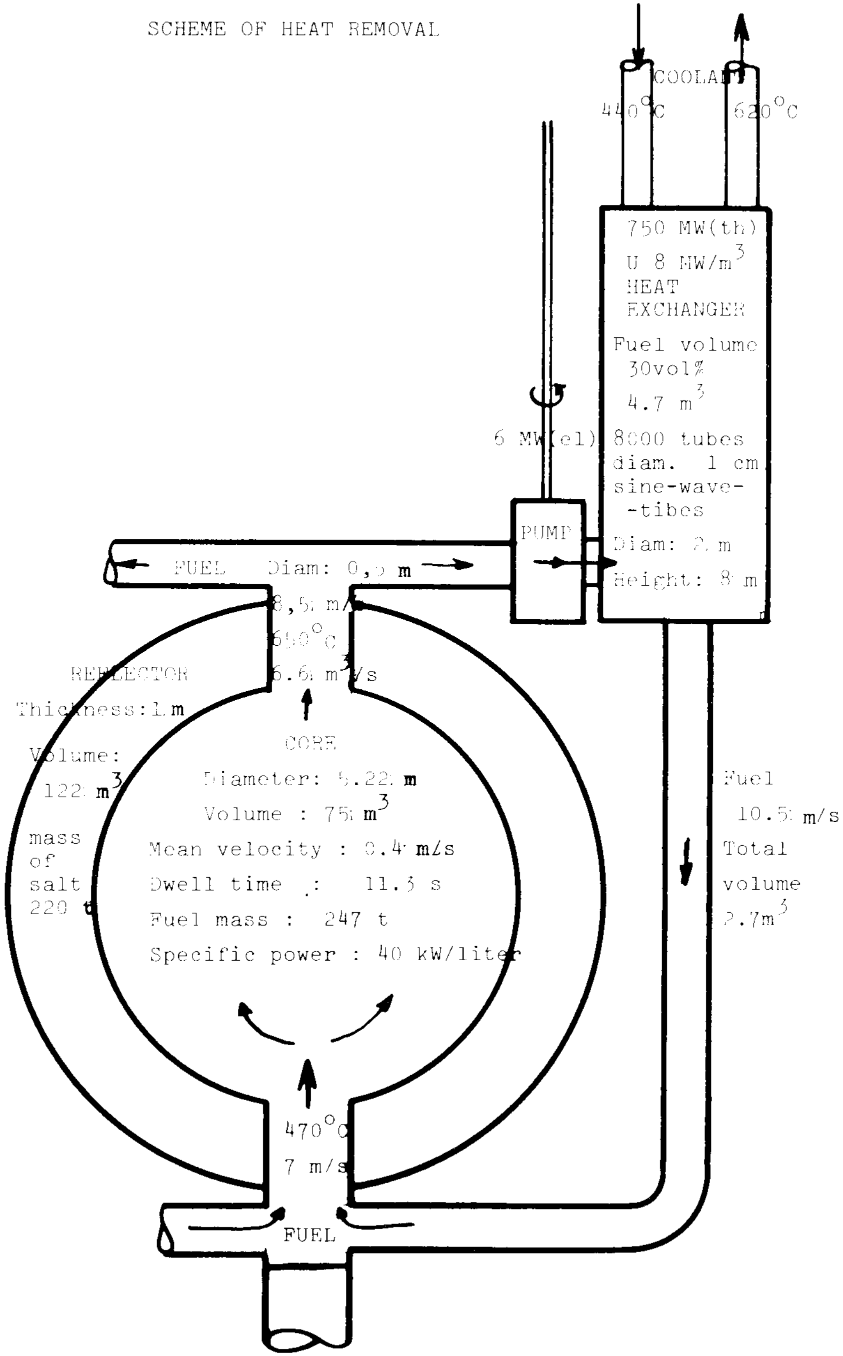
Table 5.4 In what form should the Fission Products be stored?

	Criteria	Positive	Negative
1	Chemical stability (radiation damage)	oxide, chloride, fluoride. metals, alloys.	carbonate, sulphate aqueous solutions
2	Low volatility	oxide chlorides	metals
3	Low pressure	solid or molten state	aqueous solutions
4	High thermal conductivity (heat transport)	metallic - solid metallic - liquid molten salt	oxide, solid compounds.
5	Easy management - storage - solidification	liquid	solid gaseous
6	Low specific power	diluted	concentrated
7	Small volume	concentrated	diluted

See also Fig. 5.6

Fig. 6.1.

SCHEME OF HEAT REMOVAL



The criteria for selection are as follows:

- the highest possible thermodynamic efficiency,
- the full use of the high temperature of the molten fuel reactor,
- minimising of the risk of chemical reactions between secondary and tertiary working agent,
- minimising the pressure in the working agent,
- use of the same circuit materials as for the secondary circuit if possible.

All these requirements can be met by using aluminium trichloride as the working fluid. This has been discussed earlier (Ref. 14, 15).

6.4 Heat Removal from the Stored Fission Products

An additional problem for this reactor arises from the continuous extraction of the fission products from the molten fuel and transfer out of the reactor containment. For this a second (small) containment must be built to house the stored 'hot' fission products.

The power produced will be of the order of some tens of megawatts (th) depending on how quickly the fission products are removed for further treatment elsewhere.

This system therefore requires a special cooling system which can be built with some simple redundancy relatively inexpensively.

The problem of the possible processing of the fission products and transportation methods have not been discussed here.

7. ACCIDENTS - PROBLEMS AND SOLUTIONS

7.1 Decay Heat: Power and Energy

The crucial problem of each fission reactor is the problem of the decay heat removal during a worst core accident.

It is known that the fission product inventory is:

- for short lived F.P. proportional to the specific power (that is to the neutron flux),
- for long lived F.P. proportional to the energy produced (that is to the neutron flux times length of operation)

To carry out a rough estimate the following simplified calculations have been made:

a) the decay heat power from a fission reactor has been taken from ANS guides (Ref. 1) (see Fig. 7.1). As a reference the decay heat from the thermal fission of ^{239}Pu has been taken,

b) for the sake of simplification the decay heat power curve has been presented as an overlapping series of six curves having half-life value of 10^8, 10^7, 10^6, 10^5 10^4, 10^3 and 10^2 seconds.

c) the decay heat of ^{239}U and ^{239}Np has been taken into account

d) the build-up curve has been constructed for an irradiation time of 500 days and 5 days (Fig. 7.2),

e) the effect of gas extraction on the volatile fission products has been assume to be as follows:
   - for the F.P. with $t_{1/2} = 10^2$s only a quarter of the total can be extracted by gas pumping,
   - for the F.P. with $t_{1/2} \geqslant 10^3$s approximately one-third can be extracted by gas pumping.

f) the effect of chemical extraction on all the long lived fission products with $t_{1/2} \geqslant 10^5$s has been estimated to be approximately 99 percent (rather optimistic),

Fig. 7.1
APPROXIMATION OF THE
DECAY HEAT POWER
(Reference: LWR; 3 GW(th); 500 days)

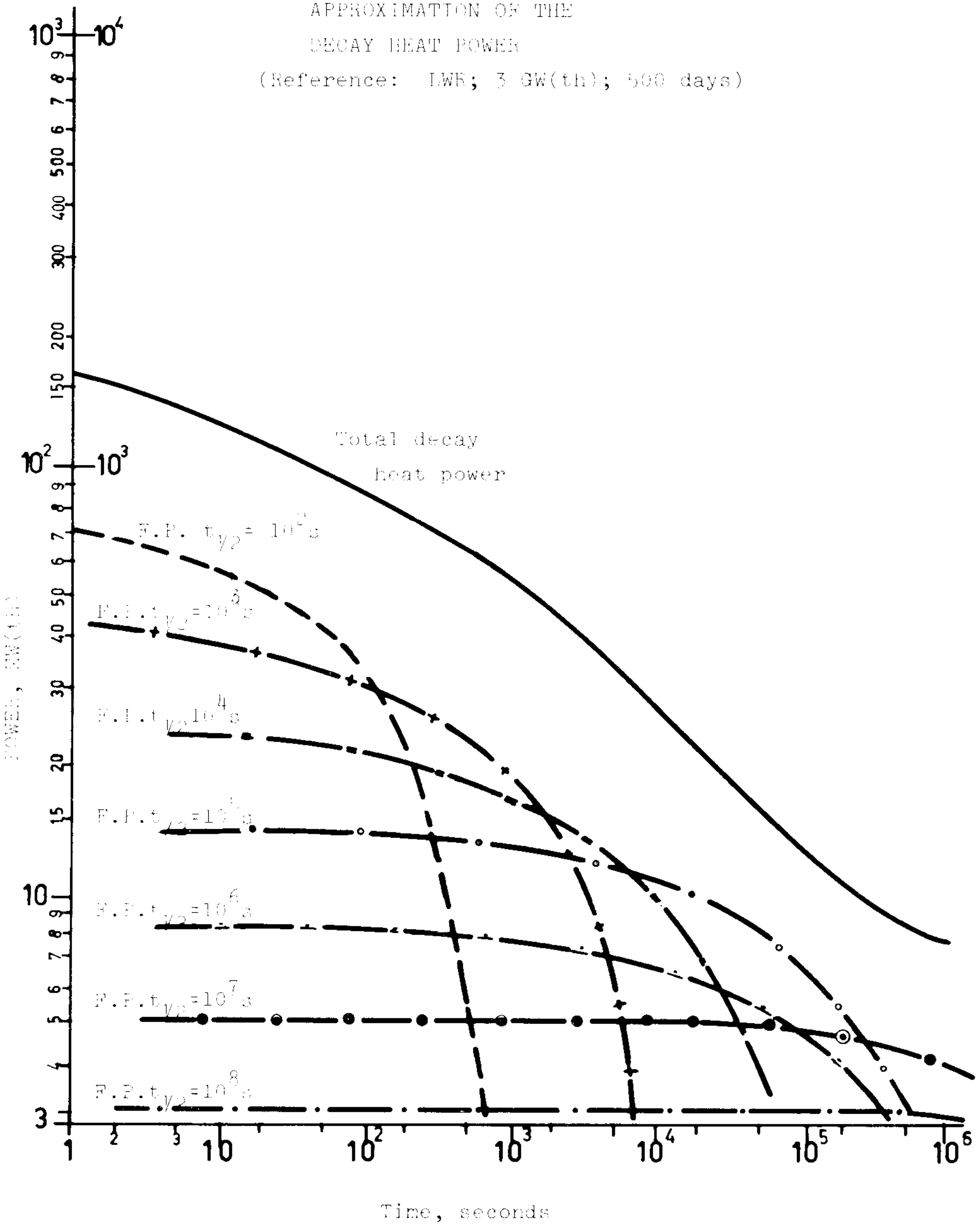
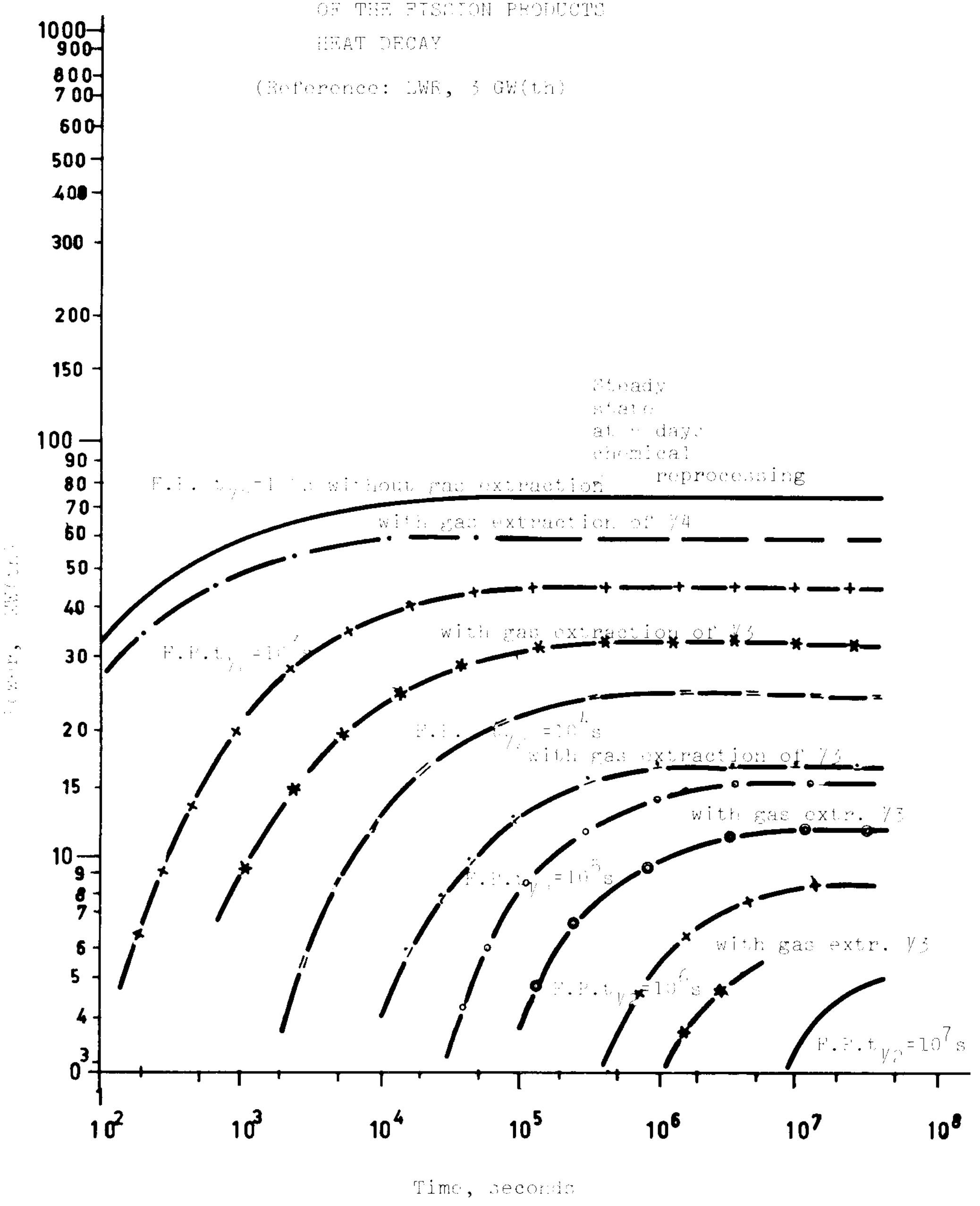


Fig. 7.2
SIMPLIFIED RECONSTRUCTION
OF THE FISSION PRODUCTS
HEAT DECAY

(Reference: LWR, 3 GW(th))



g) Figure 7.3 shows the different afterheat power curves for the SOFT reactor including all the processes and nuclides mentioned above,

h) Figure 7.4 shows the afterheat power curves and the resulting integrated heat produced as a function of time.

7.2 Worst Accident Scenarios

A number of simplified scenarios for the worst core accident for this reactor is shown in Fig. 7.5.

The calculation of the heat and power for each stage of the calculation is given in Table 7.1.

The following three points have been taken into account

a) the scram is perfect (in this reactor type not only the engineered fast shut down system, but also the inherent negative reactivity (coefficient).
All the cooling possibilities fail. The fuel in the core and in the external loop begin to heat up as a result of the decay heat.

A temperature of 950°C is reached after a time of 0.7 hours after the scram. This temperature has been postulated as the maximum 'allowable' temperature of the fuel because of:

- partial pressure of volatile components in the irradiated fuel,
- structural material properties (e.g. molybdenum alloys for the core vessel),
- management of molten fuel.

The end of this step is shown in Fig.7.4 and Table 7.1 as point A.

b) an interim mechanism has been postulated in that the reflector material being composed of chlorides of e.g. NaCl, $CaCl_2$, MgCl etc.

Fig. 7.3

AFTER HEAT DECAY CURVE
INCLUDING DECAY OF ^{239}U AND ^{239}Np

(with gas purging and with
chemical reprocessing)

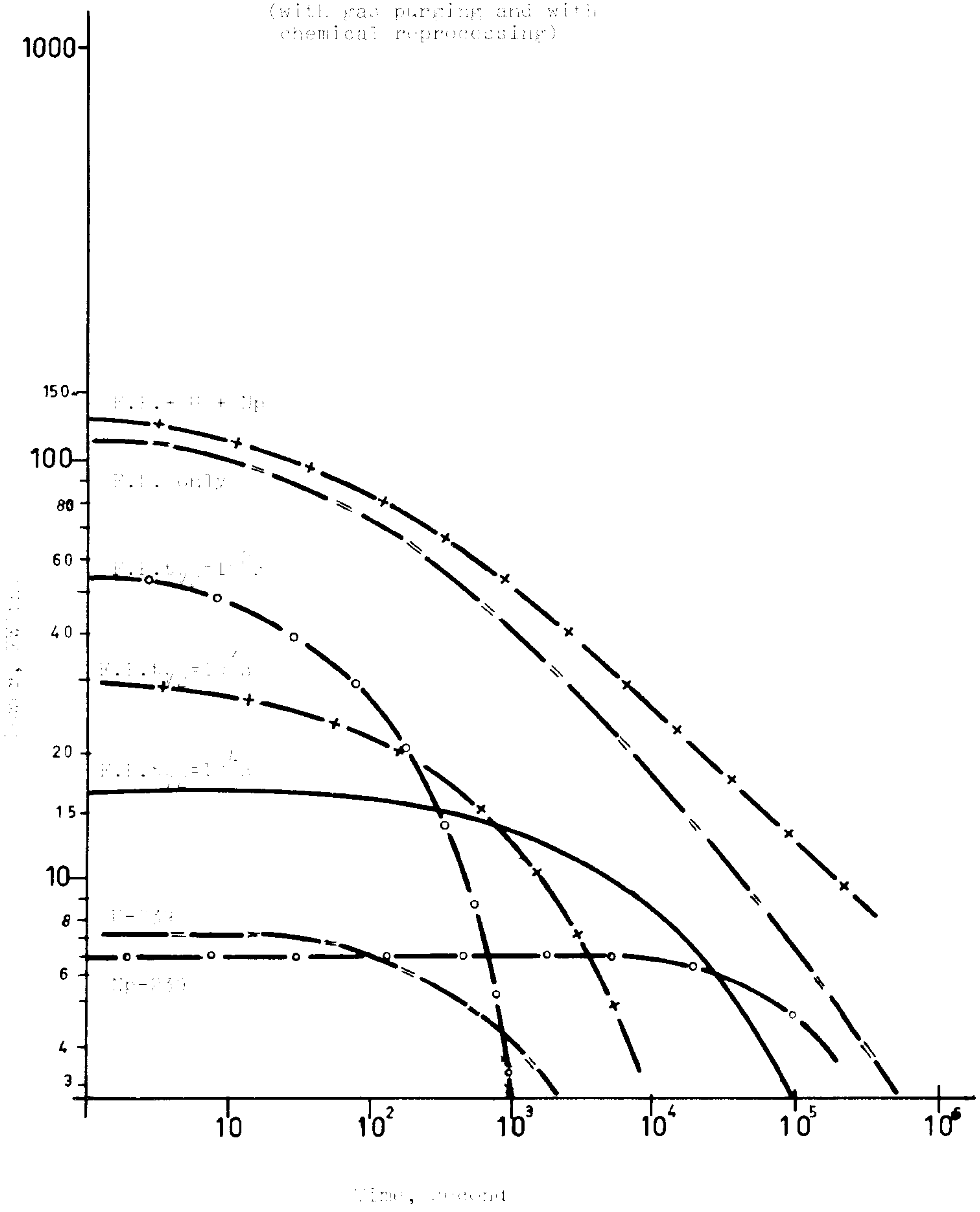


Fig. 7.4
POWER AND INTEGRATED HEAT
FROM RADIOACTIVE DECAY OF
FISSION PRODUCTS AND
ACTINIDES IN SOFT-REACTOR

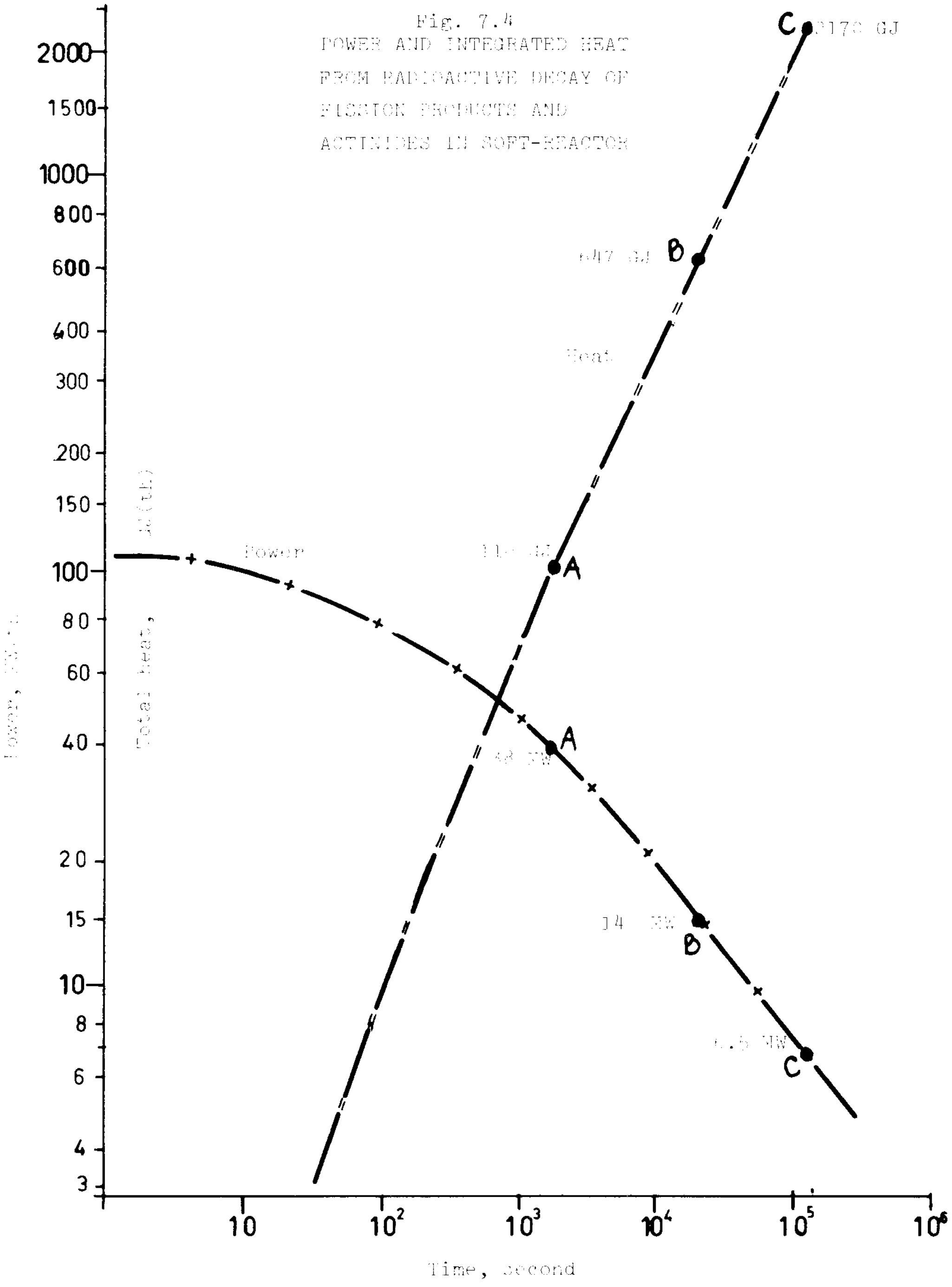


Table 7.1 Scenario of the worst accidents

	Stage	Calculation	Heat in this stage (GJ)	Cumulative heat from start (GJ)	Elapsed time from start (h)	Decay heat power at given time (MW(th))
A	After scram all cooling systems fail. Self heating of fuel	Mass of fuel 353t Specific heat 0.8MJ/t.K Mean fuel temperature at start $\frac{470+650}{2}$ Mean fuel temperature at end 950°C	110	110	0.7	38
B	Reflector begins to melt	Mass of reflector 660t Specific heat ∿1.0MJ/t·K Heat of fusion 0.5GJ/t Temperature start 470°C Temperature end 700°C	540	650	7	14
C	Fuel is drained to the core catcher (reflector is stationary and is neglected)	Fuel temperature 950°C Fuel mass 353t Catcher mass 2000t Catcher temperature start 20°C Catcher temperature end 600°C Specific heat 0.9 Fusion heat 0.5	2060	2170	42	6.5
D	Guillotine	Fuel mass 353t Mean fuel temperature at start 560°C Catcher mass 2000t Catcher temperature 20°C Heat of fusion 0.5GJ/t Core catcher final temperature 600°C				
		Fuel heat loss: 353x0.8(950-600)	100			
		Salt heating and melting 2000 (1.0x(600-20) + 0.5			42	6.5
		Balance of heat		2170		

Fig. 7.5
SCENARIO OF THE ACCIDENTS

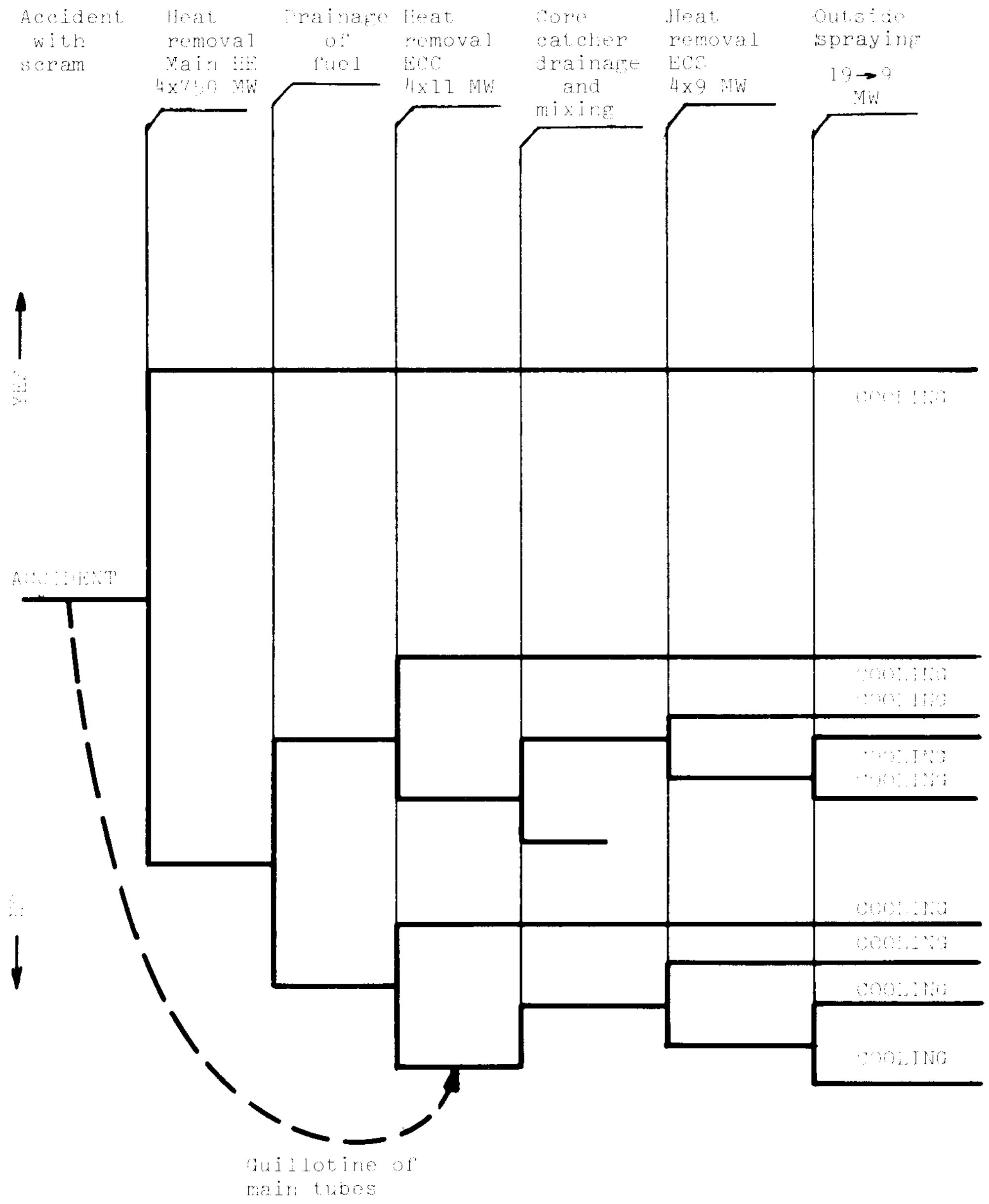
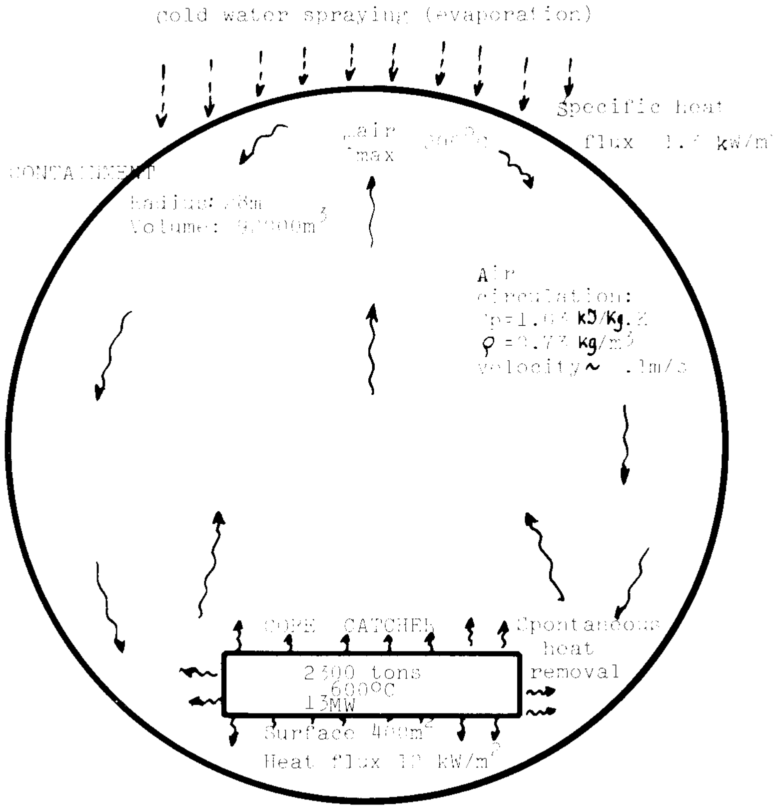


Fig. 7.6
SCHEME OF THE HEAT
TRANSPORT PROCESSES
IN THE CONTAINMENT



and as a result of the temperature increase from 470°C to 700°C and then 950°C, also heat up and melts. (Note the problem of heat transfer from the core to the reflector has not been discussed). Point B on Fig. 7.4 and Table 7.1,

c) the drainage to the dump tank is negligible,

d) the fuel at 950°C drains to the core catcher filled with ∿2000 tons of solid chlorides with a melting point of approximately 600°C. (Point C: Fig. 7.4 and Table 7.1)

7.3 Spontaneous Cooling Processes in the Containment

In this scenario it has been assumed that all cooling systems (including all redundancy) and all emergency cooling systems of the core of the dump-tank of the core catcher and of the internal containment fail.

Figure 7.6 shows the principle layout of the reactor after drainage of the fuel to the core catcher. The transport of heat from the molten fuel and the core catcher mixture is as follows:

a) the molten mass of∿2350 tons reaches the permissible temperature of 600°C after 42 hours from the start of the accident (and scram). At this moment the decay heat is approximately 6.5 MW(th).

b) the core catcher has the following characteristics
   - the upper part free molten salt and a temperature of 600°C (873 K).
   - the lower part of steel at a temperature of 450°C (723 K)

   The heat losses due to infra-red emission are:
   - for the molten salt surface (ε = emissivity)
     $Q^1 = 400 m^2 \times (\varepsilon = 0.5) \times (\sigma = 5.67 \times 10^{-8}\ J/(m^2K^4)) \times (873)^4 = 6.5$ MW

- for the steel structure

  $Q'' = 800\ m^2 \times (\varepsilon = 0.8) \times (\sigma) \times (723^4) = 9.9\ MW$

  The total heat emission due to infra-red radiation allows the removal of all the decay heat once this has dropped below 10 MW.

c) this heat has to be removed from the air and the following parameters have been taken:

Temperature (^oC)	Heat Capacity (K.Joule/kg)	Density (kg/m^3)	Pressure (bar)	Effective density
50	1.01	1.16	1.1	1.27
200	1.03	0.74	1.7	1.26

For the simplified calculation of the heat transfer coefficient for natural correction in air the following simplified relationship has been used (in BTU, ft, ^oF)

$h = 0.22\ \Delta t^{1/3}$ (according Kern 1950: $h_c = 0.38\ \Delta t^{1/4}$)

for $\Delta t = 100^oC$ the heat transfer coefficient :

$h = 7\ W \cdot m^{-2} K^{-1}$ (from Kern 1950 : $h = 7.9\ W \cdot m^{-2} K^{-1}$)

for the bulk temperature difference :

$\Delta T = (T_{air} - T_{steel}) = 200 - 30 = 170^oC$

the heat flux will be :

$\dot{H}_{air} = 7 x 170 = 1.2\ kW/m^2$

The desired heat removal from the outside wall of the containment (dia 56m, for half of surface)

$$\dot{H}_{decay\ heat} = \frac{6.5\ MW}{5000\ m^2} = 1.3\ kW/m^2$$

d) the air as a heat transport medium :-

Heat capacity of 1 m^3 air $= 1.02 x (1.16) x (200-50)$

$= 177\ KJ$

7.4 The Prompt Critical Scenario

Since in this study only steady state calculations have been made the problem of the prompt critical excursion is not discussed seriously.

However a rather naive attempt is made to obtain a rough idea of the magnitude of events under such a scenario as follows:

a) the best known data for a prompt critical accident for the liquid metal cooled fast breeder SNR-300 has been taken as the reference (Ref. 11)

b) where the SOFT reactor has significantly different characteristics such as:

temperature of fuel during normal operation:

SNR - 1154°C
SOFT - 650°C

a heat capacity of total fuel inventory for $\Delta T = 1000^{\circ}C$:

SNR - 0.21 GJ/100 K
SOFT - 28.2 GJ/100 K

\- then the values for the SOFT reactor have been taken.

Table 7.2 shows the calculations.

The reactor proposed here has therefore the following characteristics:

a) the continuous removal of fission products and the adjustment of the fissile/fertile inventory continuously during operation. This minimises the amount of excess reactivity which must be compensated for by the control rods, which reduces the risk of rapid control rod withdrawal causing an uncontrolled excession,

Table 7.2 Prompt Critical Accident

*) assumed by analogy.

Parameter	Method of calculation	Unit	SNR-300	SOFT
Reference	-	-	KfK-2845 Dec.1979	This paper
Nominal power (Po)	-	GW(th)	0.76	3
Accident scenario	-	-	F8 with compaction	
Ramp	KfK ; $\dot{\rho}$	\$/s	80	80*
Reactor period	"	ms	3.2	3.2*
Duration of excursion (t^{exc})	"	ms	23.3	23.3*
Increase of peak power	$e^{23.3/3.2}$	$P^{max}/_{Po}$	1500	1500*
Increase of average power	roughly estimated	$P^{av}/_{Po}$	342	342*
Total energy	$(P^{av} \times t^{exc})$	$P^{o}s$	7.97	7.97*
Total energy relative to solidus (Q)	KfK	GJ(th)	6.1	23.9
Mass of molten fuel (M)	KfK	t(percent)	6.5(99%)	225(100%)
Heat capacity (C_p)	literature	MJ/kg.K	0.33	0.80
Temperature increase (ΔT)	$Q/(M^{melt} \times Cp)$	K	2840	133
Operating fuel temperature (T^{nor})	KfK(1684)	°C	1154	650
Average fuel temperature after accident	$(T^{nor} + \Delta T)$	°C	3994	783
Fuel temperature (accident)	KfK	°C	3920	
Fuel temperature maximum (accident)	KfK	°C	5060	
Pressure after accident	KfK	bar	90	3

	Reference	Unit	SNR-300	SOFT
Compaction		-	possible	??
	Wirtz, p.105			
Doppler constant	γ	-	0.004	
Characteristic time for excursion (prompt generation time)	λ	s	4×10^{-7}	
Delayed neutrons	β	-	0.003	0.00
Neutron lifetime	l	s	10^{-7} 10^{-5}	
	$\alpha = \frac{\kappa(1-\beta)-1}{l}$	s^{-1}	2400-3600	
Reactivity	k			
Power at prompt critical	$P(t)=P_o \exp(\frac{\rho \cdot t^2}{2 \cdot \Lambda})$			
Total energy release				
Fuel in core time	KfK	day	441	5
Maximum burn-up	KfK	MWd/t	69000	not applicable due to continuous reprocessing
Fluence	KfK	n/cm^2	13×10^{23}	
Flux		$n/cm^2 s$	$(\phi_{av}=3.4 \times 10^{15})$	$(\phi_{av}=5 \times 10^{15})$

b) the temperature reactivity coefficient $\frac{\partial K}{K \partial T}$ is dependant on two effects:
The removal of the fuel out of the critical zone due to density decrease (temperature coefficient of expansion equals $\sim 1 \mathrm{x} 10^{3}$/K) and secondly the increased leakage of neutrons,

c) the inflow of cooler fuel occurs with time constants of at least several seconds and does not strongly affect the control characteristics,

d) under normal operation the core contains $\sim 0.2\%$ of helium bubbles. Changes in gas flow changes the average density of the molten fuel. A depressurization of the fuel system which would expand the bubbles by a factor 2 to 3 would cause a reactivity decrease,

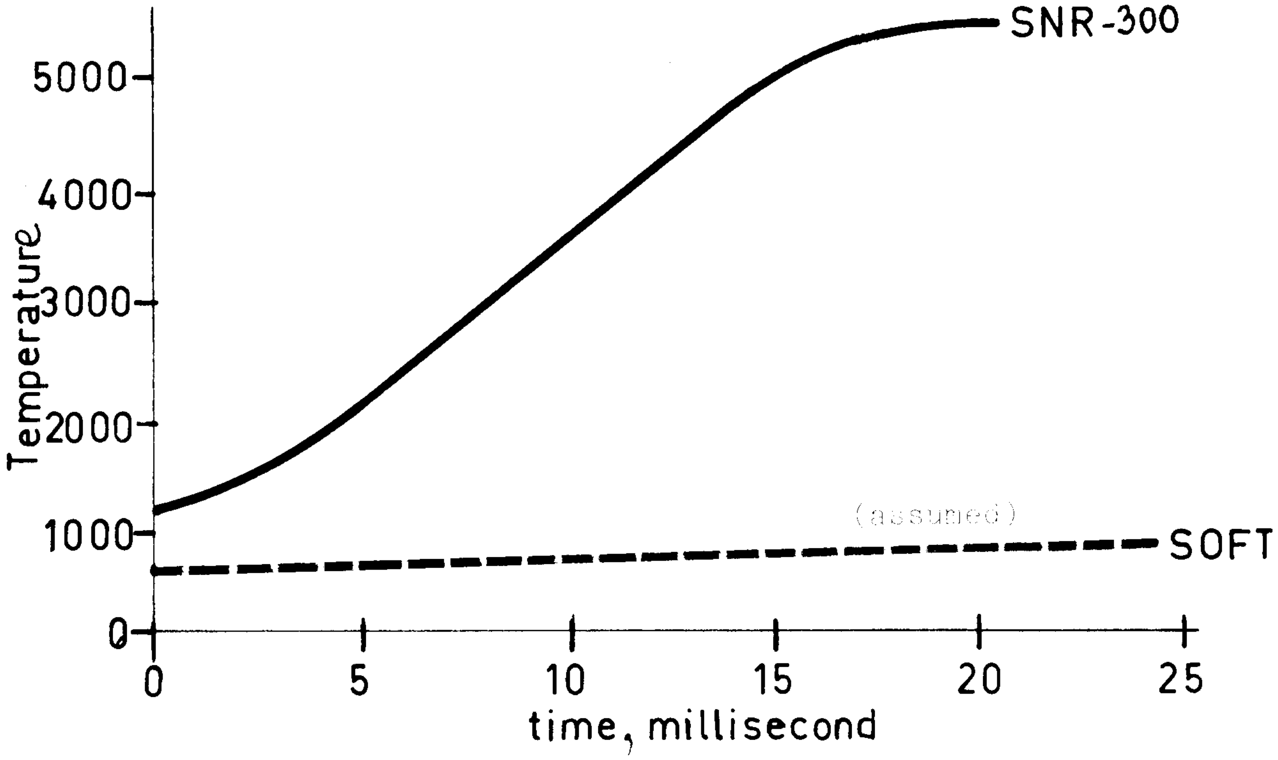
e) the voiding by local boiling in this type of homogeneous core is not credible even without pumping,

f) the fissile material in the on-line processing plant amounts to less than 1% of the reactor inventory. If all this amount could be returned to the reactor within one minute the excess reactivity would be increased by 0.007% per second (by rough estimation),

g) the worst credible event occurs probably due to a breach of the primary sytem boundary. This results in a release of large amounts of irradiated fuel - even the total amount into the containment building. The rather simple core catcher in the form of ~ 2000 tons of salt (see Fig. 3.2) is sufficient to control this accident,

h) a peculiar property of this reactor is the possibility of inhomogeneity in the fissile material ($PuCl_3$) for example due to hydrolysis and/or reaction with oxygen resulting in the formation of insoluble PuO_2 . The probability of such an event is very small since no water is used in the entire system.

Fig 7.7
Temperature increase during prompt criticality



8. CONCLUSIONS.

The reactor concept has the following strong points:

- the steady state operating inventory of potentially harmful fission products is significantly reduced,
- high thermal stability against excusion,
- removal of the decay heat is guaranteed even in case of a failure of all emergency cooling systems provided the outside containment spraying system works, for a power of less than 0.5 MW.

however;

- the technology of molten salt reactors is not yet sufficiently developed,
- the corrosion problems will probably be difficult to solve,
- the amount of Pu present in the reactor is very high, (but not optimized)

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