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ABSTRACT

A reference design for a 200 kWe Space Nuclear Power System has been developed by the CNES and CEA Agencies of the French Government in order to assess within a first study phase running from mid 1984 to mid 1986, the key feasibility issues and the development cost of a Space Power System compatible with the version of the European launcher (ARIANE V), that will be available after 1995, and with adequate power range and lifetime performances for the missions considered at that time. The heat from a fast spectrum lithium cooled reactor is converted by a turboelectric system, selected for its technological readiness and for its advantage over thermionics and thermoelectricity, of minimizing the total mass of 100 to 300 kWe power systems, considering the available radiator area afforded by the specific ARIANE V geometrical features. A heat pipe radiator is preferred to an equivalent gas cooled system, for the increased reliability brought by the large number of independent cooling elements. The successive topics addressed in the paper, include a description of the system main components and steady state operating conditions, and the present views about the start up procedure and the reactor control.

INTRODUCTION

The present European ARIANE space program will expand after 1995 in the development of the large ARIANE V launch vehicle. Considering, that the range of power needs (50 to 400 kWe) and operation times required for the space missions planned after the year 2000, are relevant to a nuclear power system, the French "Centre National d'Etudes Spatiales" (CNES) invited in 1983 the "Commissariat à l'Energie Atomique" (CEA) to undertake preliminary studies on space power systems[1,6].

The purpose of the present two years phase (mid 1984-mid 1986) is to identify key technologies for a space generator within the power range of interest, and to estimate the development cost of such a project to be examined for commitment in 1986. This work mainly consists in the feasibility and cost assessment of a reference 200 kWe turboelectric space generator illustrated in figure 12, that basically consists of a fast neutron spectrum Lithium cooled reactor[2], of a Brayton conversion loop[3] and of a heat pipe radiator[4]. This paper successively reports a description of the system main components and steady state operating conditions, and the present views about the start up procedure and the reactor control.

TURBOELECTRIC CONVERSION SYSTEM

The incentive to a turboelectric conversion system, proceeds from the extended experience in turbo-machinery technology, from the availability with moderate Research and Development effort, of adequate Brayton units for the project, and also from the attractive specific mass of a power system equipped with this converter in comparison with thermionics and thermoelectricity, considering the available radiator area (180 m^2) afforded by the specific ARIANE V geometrical features.

The use of four Brayton loops working in parallel at half nominal rated power between common primary and secondary heat exchangers is intended to permit a back up operation at quasi nominal power in case of conversion units partial unavailability.

A parametric survey[3] of the Brayton cycle main parameters was performed with a view to optimizing the integration of the energy conversion system and its associated radiator, in the launching bay of ARIANE V ; this converged on the reference He-Xe cycle illustrated in Figure 1 and on the

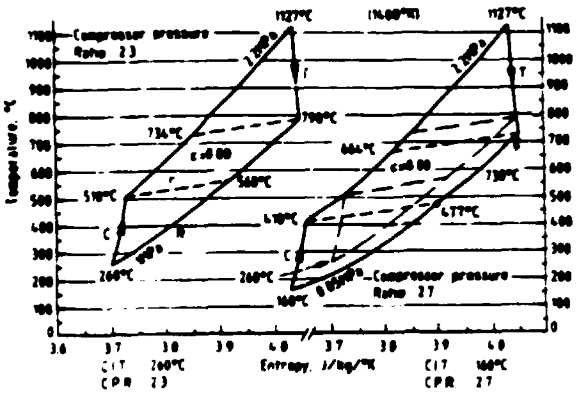
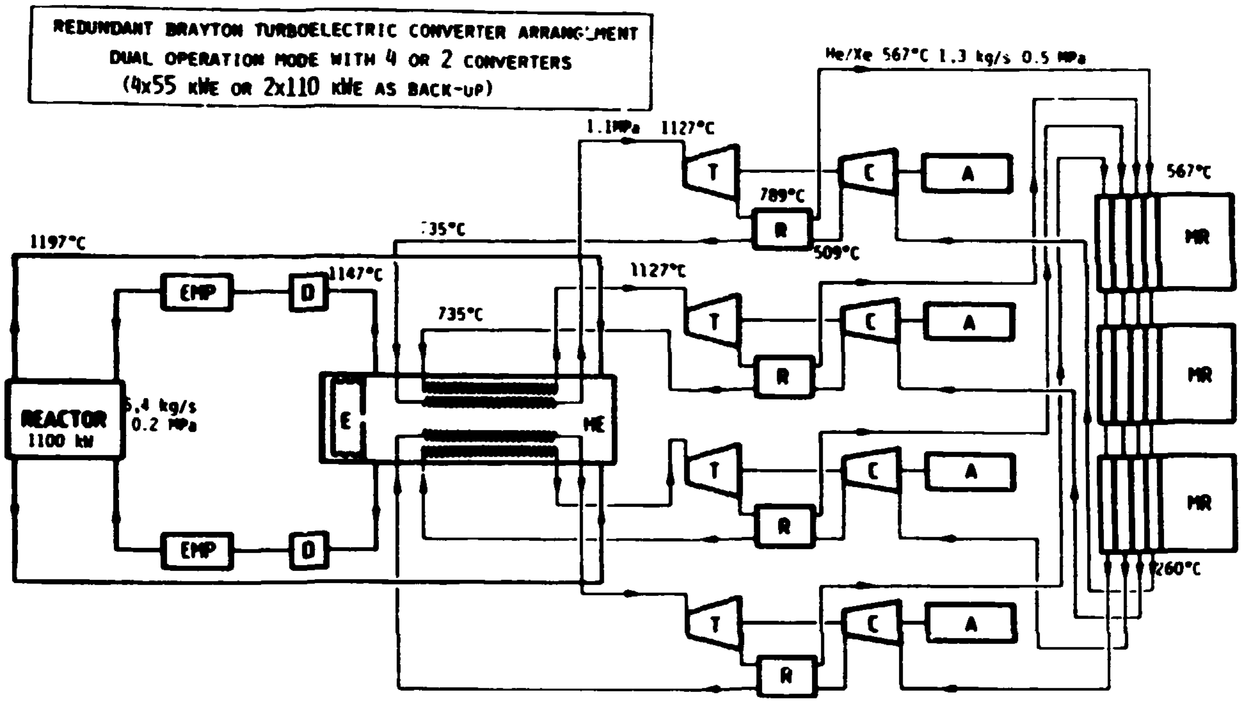


Figure 1 Temperature - Entropy Diagram for the Reference Brayton Cycle and an Alternative Option

corresponding flow diagram indicated in Figure 2 for nominal operating conditions. With the initial assumptions of a radiator area of 135 m^2 and of a turbine inlet temperature of 1127 °C (1400 °K), a moderate recuperator effectiveness was adopted ($\leq$0.80) and the compressor inlet temperature and pressure ratio were respectively fixed at 260 °C and 2.3.

From this design point, the benefit of a possible extension of the radiator area up to the theoretical limit of 180 m^2, was investigated[3] with a view to decreasing either the heat source or the radiator temperatures, as permitted by the improved cycle efficency, and to thus relaxing the demand upon high temperature technology. Even though not brought into effect for the reference project, the potential benefits expected from an extended radiator area and from improved



A : ALTERNATOR (55 kWe x 4)
T : TURBINE (He-Xe, 1.3 kg/s x 4 1.1 MPa)
C : COMPRESSOR (CPR ~ 2.3)
R : RECUPERATOR (590 kWth, η ~ 0.8)
EMP : ELECTROMAGNETIC PUMP
D : FLOWMETER (Lithium, 2.7 kg/s x 2 0.2 MPa)
E : LITHIUM EXPANSION DEVICE
MR : MAIN RADIATOR (820 kWth/140 m²)
HE : HEAT EXCHANGER (1050 kWth)

Figure 2 Redundant Brayton Turboelectric Converter Arrangement in the Reference Project.

versions of the Brayton cycle[5], are considered as possible improvements liable to relax some of the present design constraints.

HEAT PIPE COOLED RADIATOR

As illustrated in Figure 3, the spacecraft geometrical features make an area of 180 m^2 theoretically available for the radiator. Thereof, 135 m^2 only are presently allotted to the main radiator, in order to save a sufficient area for the payload and auxiliary radiators and to be consistent with the concern of addressing high temperature technologies (up to 1400 °K) in the reference project.

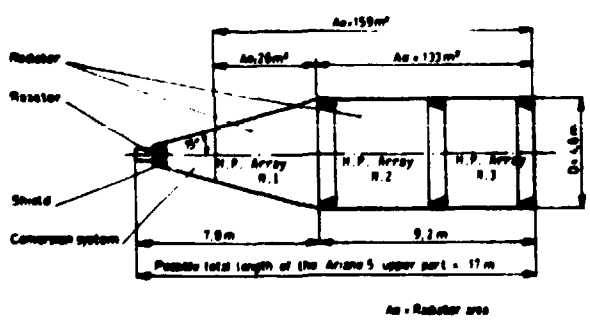


Figure 3 ARIANE 5 Spacecraft Dimensions and Radiators Areas of the Reference Project.

Reliability considerations justify the preference given to a heat pipe radiator over the equivalent gas cooled system, for the redundancy brought by 300 independent cooling elements compared with that of 4 independent cooling loops ; the gas cooled system is however recognized to simplify the radiator design and to afford a weight saving of about 250 kg out of 1400 kg.

The evaporator end of each heat pipe is equipped with a tight array of annular fins, that ensure an efficient heat transfer with each separate conversion loop. Even though it requires a more complex ducting and four independent toric headers, this configuration assures the total availability of the entire radiative surface for each separate conversion loop.

The aimed operating temperature range (240 to 550 °C) and individual power removal efficiency (2 to 3.5 kW per unit), fall within the capability of Mercury/Steel heat pipes of 1.8 cm in diameter fitted out with 4 arteries.

The radiative fin is manufactured in Beryllium and coated with a high emissivity material ($\varepsilon \sim 0.8$) ; it is mounted on the heat pipe so as to assure adequate protection against micrometeorite impacts. Figure 4 illustrates both finned heat pipe concepts, that were considered in the radiator optimization studies[4]. Gathering the heat pipes into 3 or 4 radiative panels, that cover the conical section behind the shield (1 or 2 arrays) and the cylindrical surface of the generator (2 arrays), is believed to realize an acceptable trade-off between the geometrical constraints of ARIANE V launching bay and the search for a reasonable heat pipe size, to be kept neither too long to avoid decreasing the thermal performances, nor too short to keep a tractable manifolding scheme of the 4 conversion loops around the evaporator section. Radiator optimization studies[4] aim at determining as a function of the number of arrays and of the selected heat pipe characteristics (fin thickness, inner diameter), the length of each panel, supposed made of identical heat pipes for

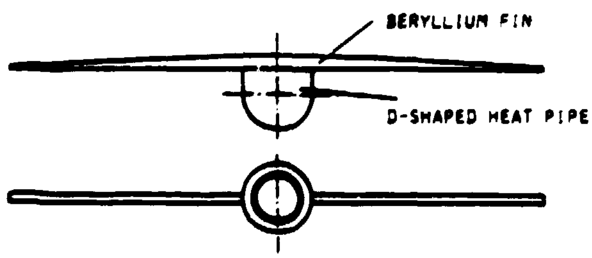


Figure 4 Considered Fin and Heat Pipe Configurations for the Radiator.

geometrical and engineering raisons, that yields the minimum specific mass (kg/radiated kW) ; the width of each individual radiative fin is then adjusted to the specific working temperature of each heat pipe.

These studies demonstrated the capability of a Mercury heat pipe radiator to meet the thermal performances required by the conversion cycle, with a satisfactory and tractable geometrical configuration.

The radiative panels and all components of the conversion system as well, are held in position by a bearing structure composed of braced tie beams rigidly supported by an internal tubular framework ; such an architecture is intented to effect an attractive trade off between weight saving and mechanic l strength.

LITHIUM COOLED FAST SPECTRUM REACTOR (1.1 MWth)

Major assets to a liquid metal cooled fast reactor, were emphasized by a preliminary review of candidate reactor concepts for this application[2]. Among these alternative options, the quasi isothermal and low pressure operation, the potential for extrapolation and the available experience from the terrestrial operation of comparable systems. Even though considered for their attractive potential for direct cycle operation and immediate readiness for start up, gas cooled pin or particle fueled reactors have not been selected for the present assessment phase, for the strain that so far investigated concepts put on the specific mass of medium power range systems. Future work in this field is however not excluded.

As input to the reflexions for resolving the critical issues of the attractive Lithium cooled reactor concept, a reference design is developed, that assumes a tight core lattice of about 1050 UO_2 sealed fuel pins and a double loop coolant supply, driven by electromagnetic pumps.

Core Configuration

Each fuel pin consists of a stack of 0.69 cm in diameter pellets at about 90 % of theoretical density, stacked in a 0.05 cm thick Mo-Re cladding of 0.8 cm in outer diameter.

The fuel pins are wrapped within a wire spacer and arranged into a triangular lattice of pitch to diameter ratio of 1.1, so as to form an active core region of 32 cm in diameter. The total pin length of 57.5 cm includes a 7.5 cm BeO reflector zone, a 32 cm active height made of fully enriched UO_2 pellets and a 15 cm plenum.

An array of 7 guide tubes (3.5 cm i.d.) inserted in the core lattice, houses a system of retractable $^{10}B_4C$ safety rods, destined to assure an adequate subcriticality margin ($k \sim 0.95$) in case of reactor submersion.

Lateral Reflector and Control System

The core is contained in a 0.4 cm thick vessel, surrounded by a BeO lateral reflector that exceeds the active height by 7.5 cm at both lower and upper ends. This reflector contains 12 rotating BeO control drums (9.5 cm in diameter) equipped with 2 cm thick and 120° wide $^{10}B_4C$ absorbing sectors (Figure 5) ; this system fulfils the

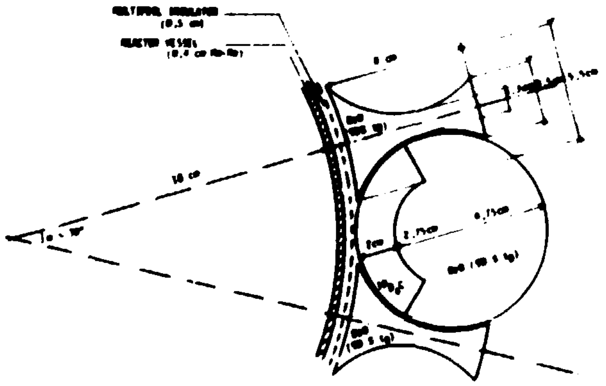


Figure 5 Generic Sector of the Lateral Reflector.

reactor reactivity and power control in normal operating conditions and is capable of assuming an adequate subcriticality margin ($k<0.95$) when the safety rods are withdrawn. Fixed reflector elements inscribed in a 8 cm thick crown, are inserted between the rotating drums. Radiative heat transfer from the reactor vessel to the passively cooled reflector is inhibited by a 0.5 cm thick multifoil insulator, that covers the inner face of the lateral reflector. The drums and reflector outer surface are covered with a high emissivity coating ($\varepsilon \sim 0.8$), that makes the radiative removal of the power deposited in the reflector (25 kW at nominal power), compatible with a moderate BeO operating temperature (<800 °C). All reflector elements are manufactured out of 90 % dense BeO and cladded with Inconel.

Each control drum rests on graphite bearings supported by props fixed to the reactor vessel ; each rotation axis, that is also made of graphite, is individually and stepwise driven by an actuator placed behind the neutron shield (1 step $\sim$ 0.1°). The control drum opening speed is restricted to less than 1 step/s to limit the consequences of an unscheduled rotation towards more reactive configurations.

The drums actuator control offers enough flexibility to assure the individual rotation of each unit or the synchronous coupling of all elements gathered in the same control drum group.

Each control drum can be instantaneously rotated back to the safe shutdown position, by a scram mechanism in the form of a spiral spring ; to that end, a declutching device is inserted in the drum axis (Figure 6). Only major accidents are intended to activate the release of this mechanism, presently considered as irreversible and destined to assure the definitive shutdown of the reactor with an adequate subcriticality margin ($k<0.95$).

Submersion Safety Rods

An adequate subcriticality margin ($k<0.95$) in case of accidental reactor submersion is assured by a system of 7 safety rods including a $^{10}B_4C$ absorber length, that covers the entire core active length and a BeO follower, that is retracted in the reactor upper structure

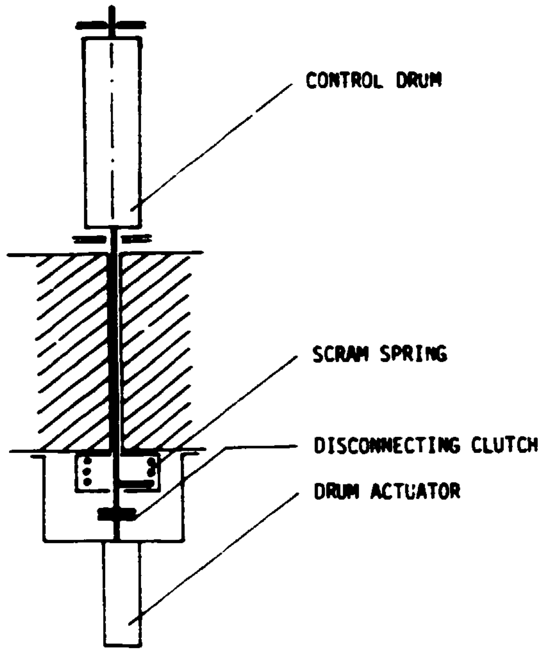


Figure 6 Principle of the Scram and Declutching Mechanisms.

during launch and then partially fills the empty tube in the configuration of operation.

In the launching configuration, the safety rods are locked in their position of maximum worth ; their diameter of 3.4 cm (including a 0.05 cm thick Mo-Re cladding) is determined, so as to assure the desired subcriticality margin (>5 %) in an hypothetical situation defined as follows :

- the core geometrical and dimensional integrity is kept
- the solid Lithium is uniformly replaced by water at 20 °C
- the lateral reflector either remains in place with all control drums locked in the configuration of maximum worth, or is dispersed and replaced by an infinitely thick water reflector.

The 7 safety rods are distributed as evenly as possible in the core section : a central rod is regularly surrounded by 6 other rods placed on a circle of 23 cm in diameter. These rods are jutting out by 15° from the control drums, so as to avoid any shadow effect on the absorbing sectors and to make all drums equivalent (Figure 7). Each rod is driven by an individual actuator placed behind the neutronic shield and can be locked in both safe and operational configurations. In order to avoid, that in case of frontal impact, a possible forward shift of the safety rods partially removes the absorbing section from the active height, the BeO follower is placed at the front of each rod and entirely fills the upper section of the guide tube up to the top of the active zone.

The safety rod withdrawal is presently considered irreversible and is executed in cold temperature condition (250-300 °K), prior to reactor start up. The non envisaged reinsertion of safety rods would offer the unique advantage to assure safe shutdown conditions (k<0.9), in case of failure or unavailability of the control drum system.

PRIMARY COOLING SYSTEM

The reactor coolant system is composed of 2 loops, each equipped with an electromagnetic pump that circulates the Lithium through the reactor and the metal/gas intermediate heat exchanger. The coolant pressure and maximum temperature at the core outlet, are respectively assigned to 0.2 MPa and to 1197 °C in normal operation at full power (1.1 MW) ; a 50 °C temperature rise within the core corresponds to a mass flow rate of 5.4 kg/s.

Except for incidents, tentatively listed below, the reactor coolant system works in nominal temperature and flow rate conditions over the aimed lifetime of 7 years at least. In case of reactor shutdown following a fault in the coolant system, the residual power is passively removed by conduction to the vessel and radiated to space by the high emissivity surface of the vessel section not covered by the lateral reflector.

Reactor Vessel

The reactor vessel has the general shape of a cylindrical shell of 0.4 cm in standart thickness, 32 cm in internal diameter and 72 cm of total height ;

it is manufactured with Mo-Re alloy. The coolant inlet and outlet plena respectively take up 10 and 7 cm at the vessel top and bottom ends. The upper vessel bottom head includes a concave membrane intended to partly accommodate the Lithium volume expansion between solid and liquid phases ; it is connected with the primary coolant loops through 2 nozzles of 4 cm in internal diameter. The lower vessel bottom is flat and serves as a fastening support to the neutron shield ; the 7 guide tubes for the safety rod system come across it ; 2 nozzles of 4 cm in internal diameter are connected with the coolant piping at the vessel lower end.

The accommodation of the linear deformations ($\sim$1 %) induced by the Lithium contraction during solidification after filling, and by its expansion during the initial coolant thawing, constitutes one of the major dimensioning difficulty of the reactor vessel and of its internal equipment.

The reactor vessel is covered with a multifoil thermal insulator on the surface area facing the lateral reflector (over 47 cm centred upon the active core height). Both lower and upper ends of the reactor vessel are covered with a high emissivity coating and exposed to space ; the corresponding thermal loss (about 50 kW at nominal temperature), is intended to passively remove the after-power with a sufficient efficiency to keep the temperature excursion after shutdown within acceptable limits (<1420 °C, that is the boiling point of Lithium under a pressure of 0.2 MPa).

The reactor vessel must be designed to withstand an overpressure of 2 MPa, likely to be induced by a tube failure in the primary heat exchanger, and to still assure the Lithium confinement in the event of such an incident. A thermal shield is fixed to the reactor top, in order to prevent the dispersion of the fuel in case of accidental reentry following an early failure during launch.

Coolant Piping and Thawing System

The selection of 2 primary coolant loops individually equipped with an electromagnetic pump, is intended to assure a certain redundancy in the coolant circulation and to maintain symmetry in the loading pattern around the axis.

The coolant interconnection piping is 4 cm in diameter (leading to an average Lithium velocity of 5 m/s) and runs around the neutron shield in helical grooves manufactured in its conical surface ; the ducting is designed so as to permit a satisfactory accommodation to dimensional changes occuring between cold and nominal temperature conditions. The pipes are entirely insulated and equipped with an electrical heat tracing. The coolant loops have no valve.

In case of primary Lithium pump failure, the Lithium circulation is inhibited in the faulted loop, either through connecting in series with each pump, a device which offers a drastic resistance to flow inversion, or through current inversion in the still valid coils of the failed electromagnetic pump. Preliminary considerations indicate that the ejection of the thermal insulator from the faulted loop, would not cause the local freezing of the Lithium in the piping.

In addition to the main interconnections between the constituents of the primary system, each coolant loop includes a small heat exchanger by-pass circuit, coiled around this component and used to thaw its Lithium content ; this circuit is also equipped with electrical heat tracing. The preliminary study of the heat exchanger thawing scenario, with various assumptions about the heat transfer from the by-pass circuit to the heat exchanger shell, finally recommends the use of 2 radiative heating ducts ($\varepsilon \sim 0.8$) of 1.5 cm in diameter and 15 m in length.

Such a device is designed to restrict the by-pass flow rate to 1 % of its nominal value in the heat exchanger during full power operation, and effects a saving of 50 to 85 % (depending on the Lithium temperature (500 to 1200 °C) in the by-pass coil) in the external energy needs to thaw the Lithium contained in the heat exchanger through electrical tracing. Considering a high Lithium temperature in the by-pass circuit is beneficial for the energy balance, as complete thawing occurs in a shorter time and the power supply needs for electromagnetic pumping consequently decrease ; however, the potential risk of thermal shocks, when the Lithium is put into circulation through the heat exchanger needs to be assessed.

Lithium Purification Units

The filling procedure of the complete primary cooling system will take place in a terrestrial facility and will include the internal cleanup of the piping system, through the prolonged Lithium circulation in the circuit with continuous purification. The Lithium solidification occurs after the circuit has been sealed and produces a volume contraction of 1.5 % likely to induce distributed or local void formation as well as deformation of the pipe walls ; these phenomena of utmost importance for the dimensioning of the piping, needs to be addressed and quantified with the help of an experimental program.

The primary cooling system is equipped with two auxiliary purification systems, that respectively remove corrosion products (hot trap) and Helium produced by (n,n' α T) reaction on the fully enriched Lithium 7. Centrifuged separation and permeation through a porous barrier are possible methods for the latter purification unit. About 16 appm of the Lithium 7 contained in the vessel (20 dm^3) is converted into Helium over a year of operation at nominal power (respectively 44 appm/y in the active core region, 12.5 appm/y in the axial BeO reflector and 3.5 appm/y in the coolant plena) ; this results in the production of 0.077 g of Helium per full power year, that would take up a volume of 1.2 dm^3 in the considered nominal operating conditions (coolant at 1197 °C under 0.2 MPa). The fact that the full power reactor operation over the aimed lifetime of 7 years would accumulate a volume of gas (8.2 dm^3) greater than the coolant content in the active core, stresses the need for an efficient separation and purge system. Any residual content of Lithium 6 would drastically increase the Helium production rate through the (n,α) reaction : a residual fraction of 900 appm would cause a gas production comparable with that originating from the reaction on Lithium 7.

Lithium/Gas Heat Exchanger

The heat transfer from both primary coolant loops to the 4 independent conversion loops, is assured by a single component designed for 1040 kW (leading to a power output of 220 kWe with a cycle efficiency of 21.2 %) with an average logarithmic temperature difference of 195 °C ; this heat exchanger consists of a bundle of gas containing pressure tubes immersed in Lithium. In normal operating conditions, the Lithium temperature drops from 1197 to 1147 °C with a mass flow rate of 5.4 kg/s, leading to an average velocity of 2.2 m/s in the bundle ; correspondingly, the gas temperature increases from 733 to 1127 °C under a pressure of 1.2 MPa, with a total mass flow rate of 5.2 kg/s for the 4 conversion loops, leading to gas velocity ranging from 60 to 70 m/s within the tubes. Geometrical and flow conditions impose a heat exchange coefficient of 0.08 $W/cm^2/°C$ and a pressure drop of 10 kPa within the tube bundle.

The bundle includes 330 tubes (1.2 m in length and 0.9/1.1 cm in diameter), that are arranged in a triangular lattice of pitch 1.6 cm. The heat exchanger globally takes the shape of a cylinder of 1.20 m in total length and 0.35 to 0.45 m in diameter, respectively in the central part and at both ends ; it is made of either Mo-Re or ASTAR alloy.

The intermediate heat exchanger is equipped at one end, of deformable bellows intended to accommodate the thermal expansion of the 68 kg of Lithium contained in the primary system and to keep the coolant pressure constant at the set point of 0.2 MPa. The bellows elongation is measured by a sensor, so as to detect any excessive deformation, that would result from the pressurization of the primary system caused by a leak of the secondary system in the heat exchanger.

REACTOR NUCLEAR AND THERMAL ANALYSIS

Fuel Operating Conditions

Full power operation (1.1 MWth) corresponds to an average specific power of about 10 kW/kgU, leading to a thermal output of about 1 kW per fuel pin. As radial and axial peaking factors of the power distribution remain close to 1.3 and 1.2 respectively over the lifetime, the maximum linear power density stays near 50 W/cm. The coolant temperature rise from 1147 to 1197 °C in the core, leads to cladding and fuel temperatures ranging from 1150 to 1200 °C and 1200 to 1400 °C respectively.

Over a typical lifetime of 7 years, the fuel would accumulate an average burn up of 25 000 MWd/t, with a peak value of 39 000 MWd/t ; corresponding fast fluence values amount to 7.7×10^{21} and 1.3×10^{22} n/cm^2 (E>0.9 MeV). The extrapolation from available data of the expected fission gas release rate is subject to large uncertainties ; the assumption of possible release rates in excess of 50 % poses the difficult problem of fission gas containment in sealed pins exposed to low outer pressure (0.2 MPa above the coolant) and to long term creep deformation. However, scarce available data on Mo-Re alloys and moderate fuel specific power give confidence in the capability of the selected 50 % plenum to fuel ratio, to accommodate the expected fission gas release over the aimed lifetime of 7 to 10 years.

Lateral Reflector Operating Conditions

Passive removal of the power deposited in the lateral reflector (25 kW at full power), by conduction and radiative transfer, prescribes working temperatures ranging from 620 to 650 °C on the control drums and from 550 to 615 °C on the fixed elements. If the initial reactor operation at full power occurs when the air stored in the multifoil insultator is still not completely outgased, the corresponding decrease in insulation performance would cause the reflector temperature to exceed by about 100 °C the nominal operating temperature.

During full power operation, the lateral reflector is exposed to a fast flux (E>0.9 MeV) of 0.5×10^{13} n/cm^2/s on average and of 2.1×10^{13} n/cm^2/s in the front region. Over a typical lifetime of 7 years, BeO will integrate a fast fluence of 1.1×10^{21} n/cm^2 on average, with, a peak value of 4.6×10^{21} n/cm^2 ; this corresponds respectively to the production of 1 and 4 cm^3 of Helium (STP) per cm^3 of BeO, with respective contributions of 75 % and 25 % of (n,2n) and (n,α) reactions.

Submersion Safety Rods Operating Conditions

During full power operation, the safety rods take the temperature of the surrounding environment ; they are consequently exposed to an important axial thermal gradient between the reactor (1147-1197 °C) and the neutron shield (400-700 °C). In normal operating conditions, a power of 2 to 3 W/cm^3 is generated within the BeO followers (depending on the peripheral or central position). The relatively poor thermal conductivity of the rod constituents, only induce a limited power deposition in the neutron shield (<1 kW) and the removal of the heat deposited in the rods requires a working temperature exceeding by 30 to 50 °C only, that of the guide tube.

The use of a high emissivity coating ($\varepsilon \sim 0.8$) on the rods and on the tubes inner surface is expected to limit at 1250 °C the cladding peak temperature.

The central and peripheral BeO followers are respectively exposed to fast fluences (E>0.9 MeV) of 1.1×10^{22} and 7.0×10^{21} n/cm^2, that respectively correspond to Helium productions of 10 and 6 cm^3 (STP) per cm^3 of BeO.

Management Scheme of the Reactor Control Dums

With the assumed core characteristics (active core height and diameter of

32 cm, and fully enriched Uranium content of 113 kg), the system of 12 control drums exhibits a total worth of 16.6 % and a residual worth of 1.6 % in the outermost position. A subcriticality margin of 10 % is therefore affordable, as the requisite reactivity margin for 10 years operation amounts to 6 % :

- 1.2 % to compensate temperature feedback effects between cold shutdown and nominal operation
- 2.8 % to compensate the reactivity drop with fuel depletion
- about 2 % as contingency margin to compensate unexpected reactivity changes ant to permit operation with a drum stuck in the In position.

All drums in synchronous rotation assure an average differential worth of 0.09 % per degree between the extreme In and Out positions, with a peaking factor of 1.7 in the neighbourhood of the middle position.

The definition of the control drum management scheme is under investigation, with major emphasis laid on flexibility and efficiency for all functions to be assured :

- control of the passively cooled reactor with frozen coolant
- reactor control during power increase and successive turbine runups
- reactor control at nominal power and compensation of the reactivity drop with depletion (about 0.26 % per year)
- safety actions activated by possible incidents occuring on the primary system or on the conversion loops.

One of the considered options is illustrated as an example in Fig. 7 and 8 :

- the control drums are gathered into 4 groups of 3 units, evenly distributed at the core periphery
- all drums of a single group are rotated in a synchronous motion
- the rotating groups follow on from each other in the order B, D, A, C with a constant phase difference of $\pi/2$ between consecutive groups.

The proposed control drum management scheme, assures an about constant differential worth ($\sim$0.05 % per degree) over the range of configurations encountered during normal operation ($0.90 < k < 1.05$). The radial

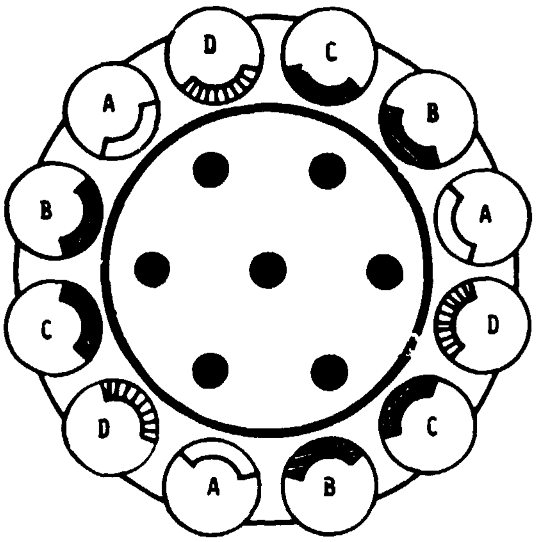


Figure 7 Distribution of Safety Rods in the Core and Management Scheme of Control Drums Based on 4 Groups of 3 Units.

peaking factor of power distribution is a quasi linear function of the control worth ; it increases from 1.25 to 1.48 when the negative reactivity insertion rises from 1.6 % (residual worth in the Out position) to 18.2 % (total worth in the In position). Criticality would correspond to a radial peaking factor of 1.30 at the beginning of life.

Alternative Subcriticality Control in Case of Reactor Submersion

The increased complexity brought by a safety rod system to the reactor and shield design, as well as the sensitivity of the subcriticality margin to possible core lattice deformations, justify the investigation of alternative subcriticality control systems based on permanent poisoning of UO_2 by oxides of the rare earth system. Permanent fuel poisoning offers the advantage of inherent safety as the reactivity of the submerged reactor configuration is little sensitive to the water distribution within the lattice, when the absorber is mixed with the

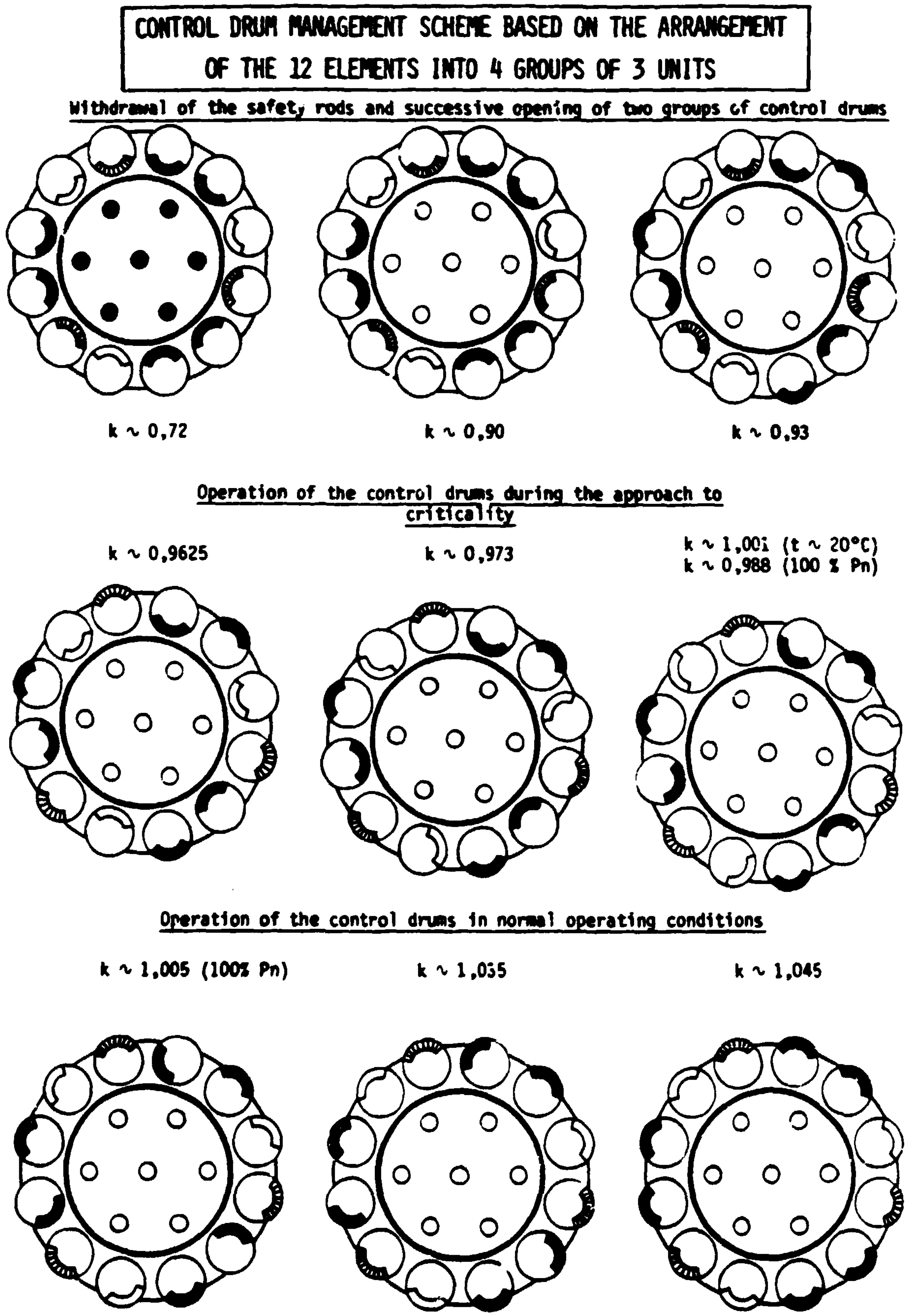


Figure 8 Example of Control Drum Management Scheme based on the Arrangement of the 12 Elements into 4 Groups of 3 Units.

fissile material. In return, the decrease in the fuel filling factor and the poison residual worth in normal operating conditions, impose a reactor mass penalty, that should be kept reasonable in comparison with that of the reference safety rod system. Reduction of the fuel filling factor is minimized by considering rare earth oxides, that are fully enriched in the isotope featured by the greatest absorbing power ; this is intended to achieve the largest difference of poisoning worth between the normal and the submerged configuration, with minimum parasitic addings to the fuel. Considerations of isotopic separation feasibility and cost are of course indispensable to complete the preliminary neutronic assessment. The Table 1 below compares the various rare earth compounds, the requisite poison volume fraction and the caracteristic dimension of an orthocylindrical core, to adjust the built-in reactivity to 4.5 % at full power (BOL) and the subcriticality margin to 5 % in case of immersion with the lateral reflector replaced by water. According to this preliminary neutronic analysis, ^{151}Eu and ^{149}Sm realize the best trade-off between thermal and epithermal absorption, in the sense that the energy range of the resonant absorption is low enough to induce an acceptable penalty in normal conditions, but high enough to be really efficient in the still hard neutron spectrum of the submerged configuration. Both considered isotopes of Gadolinium clearly suffer from insufficient epithermal absorbing power, what leads to increase the poison volume fraction in the fuel and to consequently increase the core size to offset the reactivity drop associated with a lower fuel filling factor.

The estimated reactor mass penalty of 75 and 115 kg, corresponding to the permanent core poisoning by $^{151}Eu_2O_3$ and $^{149}Sm_2O_3$ does not exceed that associated with the retractable safety rod system so much, that this alternative solution should be rejected a priori ; moreover, the relative importance of this penalty is expected to decrease as the reactor is scaled to higher power outputs. On the other hand, a limited but probable increase in the neutron shield weight with the core size, is likely to moderate the attraction of this solution, as well as feasibility and cost assessment of the isotopic separation. Europium, that only possesses two natural isotopes (47.8 % ^{151}Eu and 52.2 % ^{153}Eu) and that is, with Uranium, one of the few elements for which the chemical iso-

Table 1 Dimensional Characteristics and Fuel Composition of a Core Lattice Assigned to Typical Values of the Reactivity in Normal and Abnormal Conditions, with the Use of Rare Earth Oxides as Permanent Poison.

$k \sim 1.045$ (Full Power, BOL) $k \sim 0.95$ (Submersion)	Poison Volume Fraction in Fuel (%)	Core Height/Diameter (cm)	Reactor Mass (kg)	Mass of Uranium (kg)	Mass of Poisoning Isotope (kg)
7 Retractable $^{10}B_4C$ Safety Rods	8	32.0	480	112.8	3.7
$^{157}Gd_2O_3$	21.8	40.4	835	199.7	41.5
$^{155}Gd_2O_3$	17.5	37.3	695	165.9	25.8
$^{151}Eu_2O_3$	10.1	34.9	595	148.0	11.5
$^{149}Sm_2O_3$	11.3	33.8	555	132.7	11.3

topic separation process may be effective, offers the most encouraging prospects in this respect.

Mass Optimization of the Reactor/ Shield System

Mass optimization studies of the total system under the constraints of a given radiator area (135 m^2) and of a fixed lay-out of the conversion system, led to first investigate the sensitivity of the Reactor/Shield system mass to geometrical parameters (half cone angle, core height to diameter ratio) and to design features such as the thickness of the axial BeO reflector. With this aim in view, an optimum lay-out of the neutron shield was determined for various core configurations of the same reactivity, that simultaneously meets the requisite neutron and gamma dose attenuation, and assures satisfactory temperature working conditions of the shielding materials (for instance (400-600)°C for LiH) and minimum shield mass. These calculations were carried out with maximum allowable doses at 12 meters from the reactor, of 10^{13} nvt for fast neutrons (E>1 MeV) and $5x10^5$ rad for γ photons, over a lifetime a 7 years. Special intensification of these criteria for a possible application to an Orbit Transfer Vehicle was also accounted for : fast neutrons and γ photons dose respectively restricted to 10^{12} nvt and 10^4 rad at 15 meters over a tugging period of 150 days.

The parametric study confirmed the incentive to compact reactor configurations, such as that effected by the reference design with comparable dimensions in height and diameter ; considering larger half cone angles is a weak incitation to slightly decrease the height to diameter ratio below unity, and leads to a rapid increase of the Reactor/Shield system mass. Even though marginal from the neutronic point of view, the addition of an axial reflector of BeO between the core and the shield, is found beneficial for the attenuation of the power deposition in the front part of the shield ; the satisfactory reflecting power of the core Lithium plena justifies the lack of incentive to add an axial reflector of BeO at the upper end of the core.

The mass optimization studies converged on the following shield design, that meets both attenuation factor and LiH temperature requirements :

- the general shape is a truncated cone, with a half angle of 15°. a small diameter of 84 cm and a total thickness of 36 cm
- the successive layers encountered from the reactor are 1 cm of Steel, 10 cm of B_4C, 5.5 cm of Tungsten (with a diameter restricted to 76 cm) and 20.5 cm of canned LiH within a stainless steel structure.

The proposed lay-out leads to a shield mass of 950 kg and effects a passive removal of the deposited power (7 kW) in temperature conditions compatible with the considered materials :

- the high temperature zone (550 to 750 °C) is confined to the front region made of B_4C and steel
- Tungsten and LiH work within a narrow temperature range bounded by 400/450 °C on the lateral face, 450/500 °C on the back face and 480/530 °C on the axis.

A detailed break-down of the total system mass is given in Table 2.

Table 2 Total Mass Summary (kg) of the 200 kWe ERATO System Jointly Studied by the CNES and the CEA Agencies of the French Government.

REACTOR	480
NEUTRON AND GAMMA SHIELD	950
LITHIUM/GAS HEAT EXCHANGER	250
PRIMARY COOLING SYSTEM	250
BRAYTON UNITS (4)	670
RECUPERATOR (4)	280
CONVERSION LOOP (4)	340
MAIN RADIATOR	1800
AUXILIARY RADIATOR	250
STRUCTURE	370
INSTRUMENTATION, WIRING AND POWER CONDITIONING	360
TOTAL SYSTEM MASS (kg)	6000

LAUNCH AND START UP PROCEDURE

Launch

During launch, all 12 control drums are locked in the configuration of maximum worth and the safety rod system is inserted in the core ; both systems assure a subcriticality margin of 20 %.

In case of failure in the early stage of the launch, an adequate confinement is assumed to maintain the reactor integrity during and after fallout : the reactor geometry remains unchanged and the control drums stay locked. Neutronic studies of the submerged reactor, take nevertheless into account the most pessimistic situation, in which the lateral reflector is partly or totally replaced by water, and the safety rod system is designed to assure in these conditions a subcriticality margin in excess of 5 %.

If a failure in a later stage of the launch, must activate the dispersion of the reactor content during the fallout, a disassembly system of the lateral reflector with its control drums must be provided, in order to expose the reactor vessel and its content to the maximum thermal flux.

First Reactor Divergence

The start up sequence in orbit, that is illustrated in Figures 8 and 9 begins with the withdrawal of the safety rod system and by checking the neutronic counting channels ; then, the control drums are released and 6 out of the 12 drums (groups B and D for instance) are rapidly opened up to the Out position ; keeping 6 evenly distributed drums in the initial (In) position is enough to assure a subcriticality margin of 3.5 %. From this configuration, the reactivity is gradually increased and monitored, so as to assure during the approach to criticality and the divergence, a period greater than 60 s, up to reaching a power increase rate of 10^{-4} of nominal power per second ($\sim$11 W/s), that is maintained constant during the power ascension ; a stabilization procedure is activated, when the requisite power level is attained for thawing the Lithium content of the reactor vessel. This power level is determined at the same time by the limit of temperature increase rate to avoid excessive plastic deformation of the fuel elements under the pressure of the expanding Lithium (especially at the hot spot), and by the reactor passive heat removal capability at the aimed temperature, that mainly consists of radiative heat transfer of the reactor vessel to space and to the lateral reflector, and of conductive heat transfer to the primary coolant loops and to the vessel supporting structure. A power level of 10 kW (about 1 % of no-

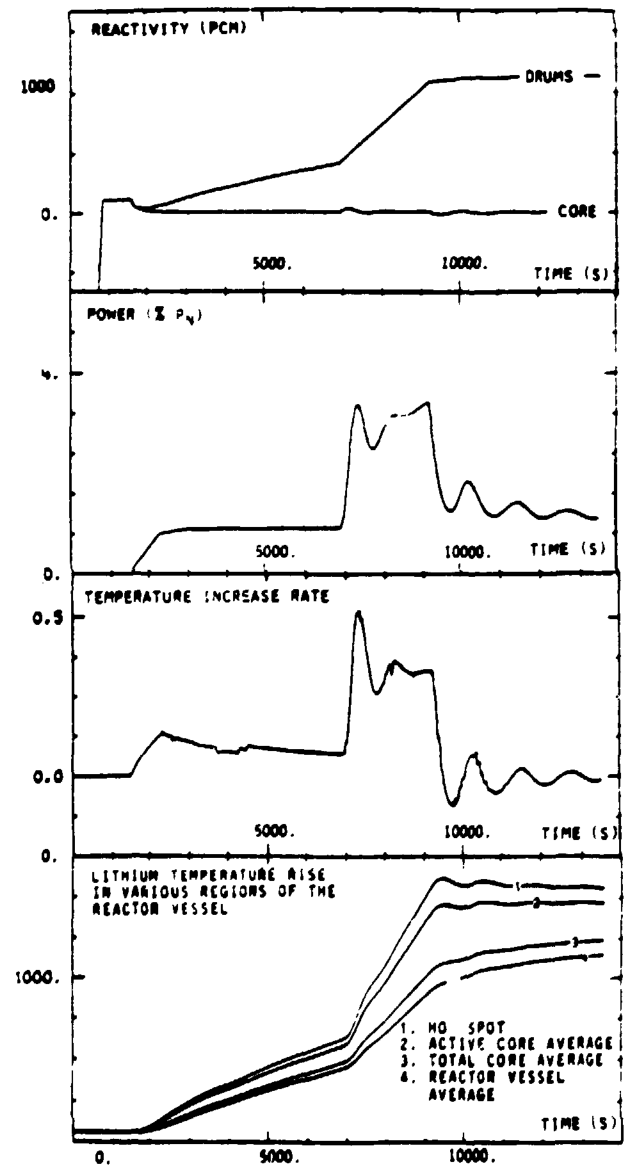


Figure 9 Thermal Response to an Ambitious Thawing Scenario.

minal) leads do a temperature increase rate of 0.05 °C/s on average and of 0.1 °C/s at the hot spot ; the temperature profile obtained at the end of the thawing transient exhibits large axial temperature gradients between the active core (900 °C) and the coolant plena at both ends of the vessel (700 °C).

Table 3 and Figure 8 illustrate the procedure of control drum operation for start up, with the considered management of the reactor control system. Figure 9 indicates the thermal response to the programmed reactivity insertion.

The sequence described in Table 3 represents an ambitious start up scenario, whose actual length will be reevaluated as the studies will gain in realism ; the reported values should be understood as orders of magnitude only.

When thawing the reactor vessel Lithium content is completed, the primary piping system and also the heat exchanger by-pass coils are heated by electrical tracing powered by an auxiliary power unit. The electromagnetic pumps are brought into service at the preheating voltage. When the Lithium is liquid in the primary coolant piping and in the heat exhanger by-pass coils, it is put into circulation by the increase of the supply voltage. The adaptation to the requisite temperature level for thawing the heat exchanger Lithium content by radiative heat transfer (t>500 °C), may require to slightly increase the reactor power level. The

Table 3 Control Drum Operation during Start up.

TIME(s)	START UP PROCEDURE	ROTATION OF CONTROL DRUMS	T_{core} (°K)	k_{eff}
	Withdrawal of safety rod system	Opening of 6 control drums out of 12 (Groups B and D)	250	0.75 0.90 0.9625
0				
	Approach to criticality until a period of 60 s is obtained	Reactivity insertion rate of 8 pcm/s Rotation of group A followed by group C with a speed of 1.5 step/s		
450			250	0.995
	Divergence with period control (60 s) until the power increase rate reaches 10^{-4} of nominal power per second	Progressive decrease of the reactivity insertion rate from 8 pcm/s to zero	250	$1<k<1.001$
650				
	Power ascension with monitored increase rate (10^{-4} Pn/s) up to requisite power level for thawing the Lithium content of the reactor vessel	Keeping the drum configuration obtained at the end of the previous sequence followed by return to criticality		
1650			250⁺	1
	Keeping the power level constant until complete thawing of the reactor Lithium content	About constant reactivity insertion rate, to compensate for the feedback effects associated with the core temperature rise (about 7×10^{-2} pcm/s (1 step/75 s))		
7500			650	1
	Power ascension up to the level corresponding to the reactor passive heat removal capability at the aimed temperature (50 kW for 1200 °C)	About constant reactivity insertion (0.3 pcm/s (1 step/15 s) for a temperature coefficient of - 1 pcm/°C)		$1<k<1.001$
10 000			1425	1
	Thawing of the Heat Exchanger Lithium content Runup of Brayton Converters Full power operation	Reactor power control system brought into service		

coolant flow rate increases as soon as the exchanger content is molten. When the temperatures are homogenized, the Lithium flow rate can gradually increase up to its set value. The Lithium is brought to the nominal working temperature (1197 °C at the core outlet) by an adequate temperature rise of the reactor power level. As the Lithium temperature increases, the thermal volumetric expansion is accommodated by deformable bellows located at one end of the heat exchanger. Maintaining the primary coolant system at nominal operating temperature causes the progressive outgasing of the multifoil insulator.

Successive Runups of the Conversion Units

The primary coolant system is operating in nominal temperature, pressure and flow rate conditions and the reactor power level just compensates for the passive thermal losses (about 50 kW at 1197 °C). The conversion system is cold and contains the suitable Helium content to reach nominal operation without make-up. Steady state, after start up of the first Brayton unit with an auxiliary power supply, corresponds to a power of about 290 kW demanded by the single conversion loop in operation (with the total radiator cooling capability available), and to the reactor operating at about 1/3 of nominal performances (340 kW).

As illustrated in Figure 10, the following control drum sequence is commanded, to adjust the reactor power level to the power demanded during the run-up of the first Brayton unit :

- a power ascension from 50 kW with an about constant increase rate of 10 % Pn per minute for 2 to 3 minutes, during which the average temperature of the primary system drops by 50 °C as a consequence of the reactor operation at a lower power level than demanded by the single conversion loop
- an operation for 2 minutes at a constant power level (440 kW), that slightly exceeds the demanded power, thus resulting in the progressive return to the temperature set values
- a stabilization of the reactor power at the equilibrium level, imposed by the core outlet temperature assigned to 1197 °C ; this is achieved under the control of the later described power regulation system, that is brought into service as soon as possible.

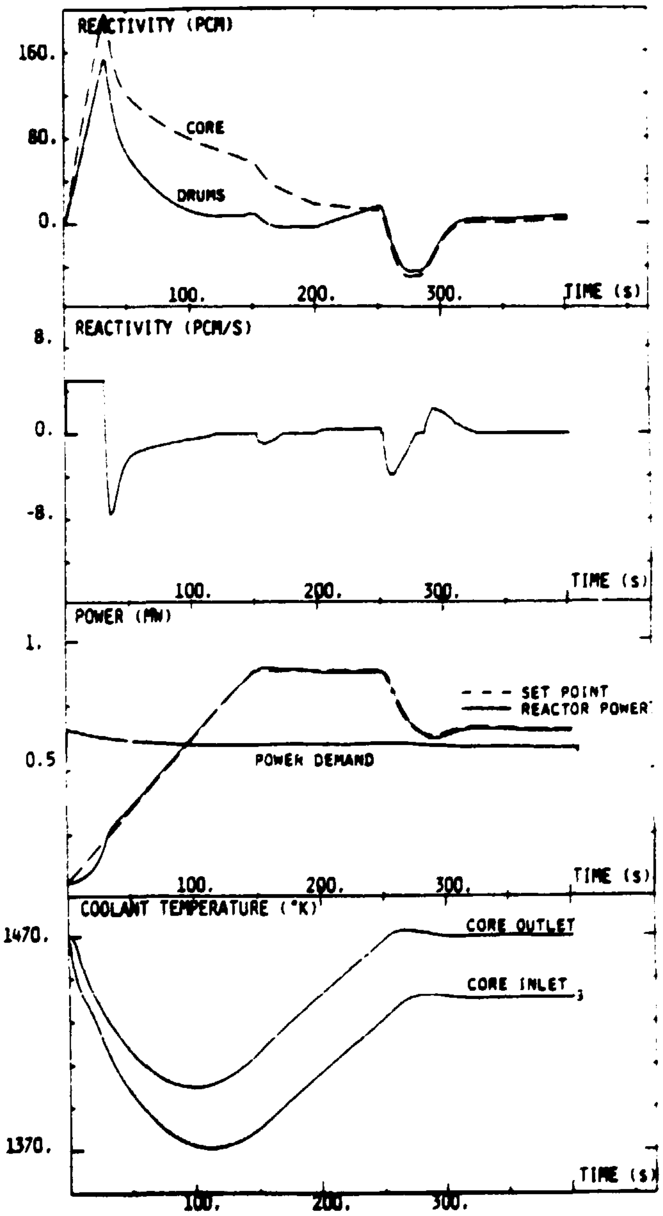


Figure 10 Reactor Power Control During the Runup Sequence of two Brayton Converters at the Same Time.

During this transient, supposed to be the fastest scenario to reach steady state after the first run-up of a Brayton unit, the requisite reactivity variation rates by the control drums, do not exceed 5 and 15 pcm/s respectively for insertion and extraction (respectively 1 and 3 steps/s with the proposed management of the control system).

The successive runups of the three other Brayton units and the simultaneous increase of the reactor power level, are performed according to the same scenario as above, except that the power supply available from the units in operation relax the dependence upon the auxiliary power unit.

After run-up and power ascension of the last conversion unit, the space power system enters the stage of full power operation, that will continue, a part from incidents, over the aimed lifetime of 7 years at least.

REACTOR POWER CONTROL AND PROTECTIVE ACTIONS

Options of the Reactor Control System

The organization of the reactor control system is based at the same time on the automatism of protective and regulative actions and on the activation of particular procedures from the ground control station, such as the thawing sequence of the reactor, the preheating of the Lithium piping and the gradual power ascension as the Brayton units are successively started up.

After stabilization at nominal power, the reactor control system is intended to maintain the reactor in a steady state ; the function of the ground control station is then limited to supervision and to periodic calibration of the instrumentation channels.

The reflexions are in progress about the principle of power control, that most efficiently stabilizes the reactor operation and damps the power and temperature swings initiated by the start-up sequence or by protective and restorative power variations in case of incident. A power control system based on the three measurements of the nuclear power, of the coolant flow rate and of the core inlet temperature offers attractive prospects in this respect : the reactivity is adjusted with a correction rate proportional to the difference between the measured power level and the ideal set-point calculated to maintain the core outlet temperature at an adjustable set value (for instance 1197 °C at full power steady state) in the measured inlet temperature and flow rate conditions. In case of unavailability of nuclear power measurements, a back-up system only based on temperature measurements may be envisaged with a probable decrease of performances.

In case of incidents, the protective automatic control circuits, bring the reactor in safe back-up operation conditions at a reduced power level ; whenever the partial or full power restoration can be envisaged, this will be commanded from the ground control station.

Low Power Back-up Operation and Safe Shutdown

The protective actions activated in case of incident of excessive drift of any essential parameter from its set point, include the following :

- Low Power Back-up Operation, that consists in a rapid decrease of reactor power, followed by a stabilization at a reduced power level, compatible with acceptable operating conditions of the possibly degraded system. As far as the system condition makes it possible, a new configuration is defined, that permits to gradually increase the reactor power up to an intermediate level or to full power. The protective power reduction is activated either by an automatic control circuit or by the ground control station
- the Emergency Shutdown, effected by the quasi instantaneous insertion of the control drum absorbing faces under the effect of the released scram springs
- the Rapid Shutdown, effected by the

commanded return of all control drums to the configuration of maximum worth within about 1 second ; the effect of this action is comparable to that of the Emergency Shutdown with a delay corresponding to the rotation time of the control elements.

The Automatic Power Reduction is achieved by a rapid negative reactivity insertion, followed by the return to criticality with a control drum configuration, that is very close to the initial one, owing to the very limited reactivity feed-back effects (- 0.5 pcm/% Pn). For instance, achieving an automatic power reduction to half of nominal power in 10 seconds and with the proposed control drum management, leads to realize a negative reactivity ramp of - 35 pcm/s (7 steps per second over 10 seconds), followed by a stabilization procedure at the back-up power level with a progressive return to criticality. In fact, the back-up power level may either depend on the nature of the detected incident or be low enough to assure acceptable operating conditions for all identified incidental situations. The actions, to be simultaneously activated to control the Lithium flow rate and the operation of the conversion units, as well as the possible procedure of partial or full power restoration, are in the process of investigation by simulation studies of each identified scenarios.

The reversible or irreversible nature of the scram spring release, is to be evaluated in function of the increase in complexity brought by a spring stretching device and by the review of the incidents, that require an emergency rather than a rapid shutdown and still authorize a power restoration at an acceptable level. The emergency shutdown is presently considered as irreversible and determines the end of the mission ; in the same way, the incident simulation studies strive to define the procedures of automatic power reduction and of rapid shutdown, so as to cover by these safety actions all the incidental situations and to release the necessity for an irreversible shutdown subject to inadvertent activating.

In fact the difference between the emergency and the rapid shutdown, with respect to the speed of negative insertion is only important for scrams activated during the reactor divergence, on the indication of an excessively short period. The necessity to maintain the emergency shutdown by spring release is to be assessed by detailed studies of reactor kinetics, as the rapid shutdown probably represents a sufficient protection against incidents likely to occur in steady state operating conditions.

Restarting the reactor and the conversion loops after shutdown requires the availability on board of a sufficient auxiliary energy source to enable the thawing of possibly frozen sections of the primary cooling system, as well as the electromagnetic pumping of the primary coolant and the run-up of the conversion units. It is important to size the auxiliary power unit to afford repeating the start up sequence of few times, in particular in the very initial stage of the system operation. It is crucial, that incidents likely to occur during the delicate start up procedure, do not activate any final shutdown but only reversible actions compatible with the resumption of the subcritical approach and divergence : the repeat of the start up procedure following an incident, that occured before thawing the heat exchanger Lithium content, only requires a marginal extra-contribution of auxiliary energy supply.

In case of loss of reactivity control by the drum system, following upon a common mode failure of the rotation control or of the actuators, the commanded ejection of the control drums also assures a safe shutdown of the reactor with a substantial subcriticality margin ($k \sim 0.8$).

Tentative Review of Incidents likely to Activate an Automatic Power Reduction

Failure of one Conversion Unit

The failure of one conversion unit (seizing up or acceleration of a turbogenerator, pierced piping), instantaneously causes the reduction of the electric power output by 25 % or by 50 % if the full power operation is already assured by 2 Brayton units only. The reactor power level is to be accordingly adjusted, either through an automatic power reduction activated by the measurement of an excessive core outlet temperature, or through a rapid action of the power control system. The simulation studies in progress, prove the principle of power regulation described earlier to be preferable to a simple reactivity correction rate proportional to the difference between the measured and the set value of the core outlet temperature (1197 °C). As illustrated in Figure 11, the preliminary analysis of the transient initiated by the instantaneous reduction of the power demanded by the converters, prove the capability of the proposed power control system to restrict the core outlet temperature excursion to + 10 °C (and about + 30 °C for the core inlet temperature) : the protective action consists in a reduction by half of the reactor power in 25 seconds followed by a stabilization at this level in less than 2 minutes and with only two moderate power swings. The requisite reactivity insertion rate for the control of this transient are less than + 5 pcm/s and - 10 pcm/s. These encouraging prospects, that a priori relax the necessity for an automatic power reduction in case of partial unavailability of the conversion system, will have to be confirmed by the detailed simulation of this incident.

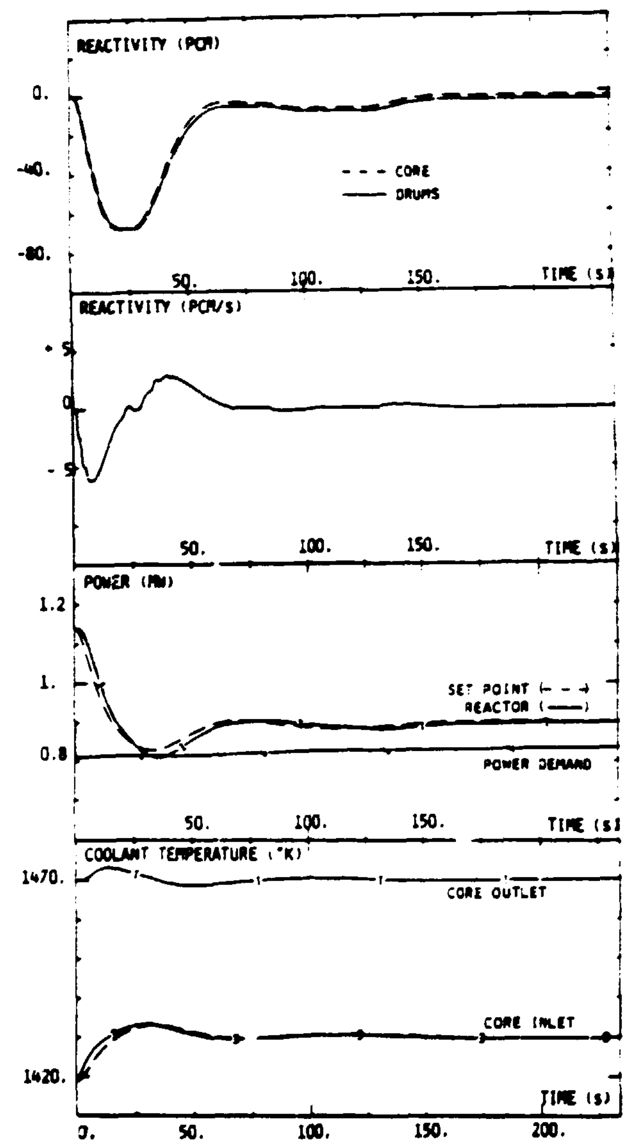


Figure 11 Thermal Response monitored by the Reactor Control System in case of Instantaneous Failure of 2 out of the 4 Converters in Operation at Full Power.

Failure of one Electromagnetic Pump

The failure of a single electromagnetic pump, induces a rapid drop of the coolant flow rate in the corresponding loop and a significant increase in the core outlet temperature, as the temperature rise across the core is amplified by a typical factor of 2. The nature and the time scale of the protective actions to be activated are a priori similar to those previously described in reaction against the failure of a conversion unit. Transient simulation studies are intended to decide, whether the control of the temperature excursion is within the capability of the power control system or requires the activation of the automatic power reduction. The flow restriction assured by a limiter device in the faulted loop, should permit the return to a power level to be defined.

Unscheduled Rotation of one Control Drum

The inadvertent rotation of a single control drum occuring at steady state induces a positive or negative reactivity variation, that is compensated by an antagonistic rotation of the other drums activated by the power control system. Any power excursion liable to result from this incident, especially if this occurs when the control system is not in service, is detected by the neutronic instrumentation and activates an automatic power reduction, when a trip threshold is overshot. The studies aim at determining the maximum authorized drum rotation speed towards the positive reactivity insertion, to keep acceptable the power excursion likely to result from this incident. A satisfactory management scheme of the control drums, should make this restricted rotation speed compatible with the reactivity variation rate required to control the reactor power level and to follow the power demand during the start up of the first Brayton unit.

The studies in progress about the definition of the control drum management and the simulation of start-up and accidental transients suggest that a rotation speed restricted to 2 pcm/s per drum is enough to permit the most rapid load following. Under these conditions, the most severe consequence of the inadvertent rotation of a single control drum would be a reactivity ramp of 2 pcm/s an average during 12.5 minutes ; this could be easily compensated by the protective action of the power control system on the other drums.

Tentative Review of Accidents likely to activate the Safe Shutdown of the Reactor

Leak in the Primary Cooling System

The detection by pressure measurements of a leak of Lithium from the primary cooling system or of gas leakage into it, activates the automatic power reduction, followed by the reactor shutdown. The investigation of the thermal transient experienced by the fuel in case of loss of coolant accident, will permit to check whether the rapid shutdown is drastic enough to assure the core integrity and the confinement of the fission products. A negative answer to this question would require to maintain an emergency shutdown device, which should be made either reversible or exempt from inadvertent activation.

Loss of the Reactor Power or Platform Stabilization Control Systems

In case of failure of crucial control systems for the reactor power regulation or for the platform stabilization, an automatic power reduction is activated, followed by the shutdown of the reactor. As above, the analysis of the accidental transients will permit to decide whether rapid shutdown is enough to fulfil the safety requirements in all situations.

CONCLUSION

The first phase of the feasibility studies for a 200 kWe Space Power System to be launched by the version of ARIANE planned for the year 1995, confirm the attractive features of the reference project, that combines the advantages of a highly efficient Brayton conversion, of a fully redundant arrangement of 4 converters with a heat pipe radiator, and of the large thermal inertia offered by a fast spectrum Lithium cooled reactor.

The Brayton conversion a priori selected for its technological readiness and for the mass advantage afforded by the high cycle efficiency, is shown to induce controllable temperature transients in the primary system, during start up and accidental failure of converters, in spite of the fact that each unit may generate up to half of the nominal power output.

The fast spectrum Lithium cooled reactor selected for its inherent compactness and mass advantage, for its efficient heat removal capability in low pressure and quasi isothermal conditions, inherently suffers from critical issues, whose impact upon the reactor start up, the design of the primary system and the reactivity control in case of submersion was assessed.

.A Lithium thawing scenario is proposed, that minimizes the needs for external energy supply and resorts on a succession of simple sequences of control drum operation intended to assure a safely monitored divergence and power ascension.

.Bellows included in the structure of the reactor vessel and of the heat exchanger are destined to accommodate the thermal expansion of the Lithium between the melting temperature and the nominal operating conditions.

.The thermal inertia afforded by the metallic coolant is found best suited to the use of a small number of efficient power converters, for the relaxation of the time scale involved in the power regulation, in case of severe and rapid transients of the secondary system such as those initiated by the run-up or the failure of a single conversion unit.

.Steps have been taken to tentatively replace the safety rod system intended to assure subcriticality in case of submersion, by permanent core poisoning, that would offer the advantage of inherent safety in compensation of a moderate increase in weight.

A control drum management scheme has been developed, that assures a quasi constant differential worth over a wide range of reactivity, so as to effect an accurate reactivity control from the beginning of the approach to criticality up to the reactor end of life, with acceptable drum rotation speeds and with an about constant correlation between the angular displacement and the reactivity variation.

Present views about the power regulation, converge on a system based on core inlet temperature, coolant flow rate and nuclear power measurements, that effects an efficient stabilization of the power swings induced by rapid variations of the power demand of the secondary system. If extendable to most conceivable incidental situations, these favourable performances demonstrated on a few typical examples, offer the attractive prospect of limiting the protective actions to a few automatic and reversible procedures such as the Automatic Power Reduction and the Rapid Shutdown.

There are significant technical challenges to take the considered conceptual design into a successful hardware system and this will be the objective of the next phase of the project, subject to commitment in 1986, to realize sound engineering solutions within acceptable budgetary and time constraints.

Acknowledgments

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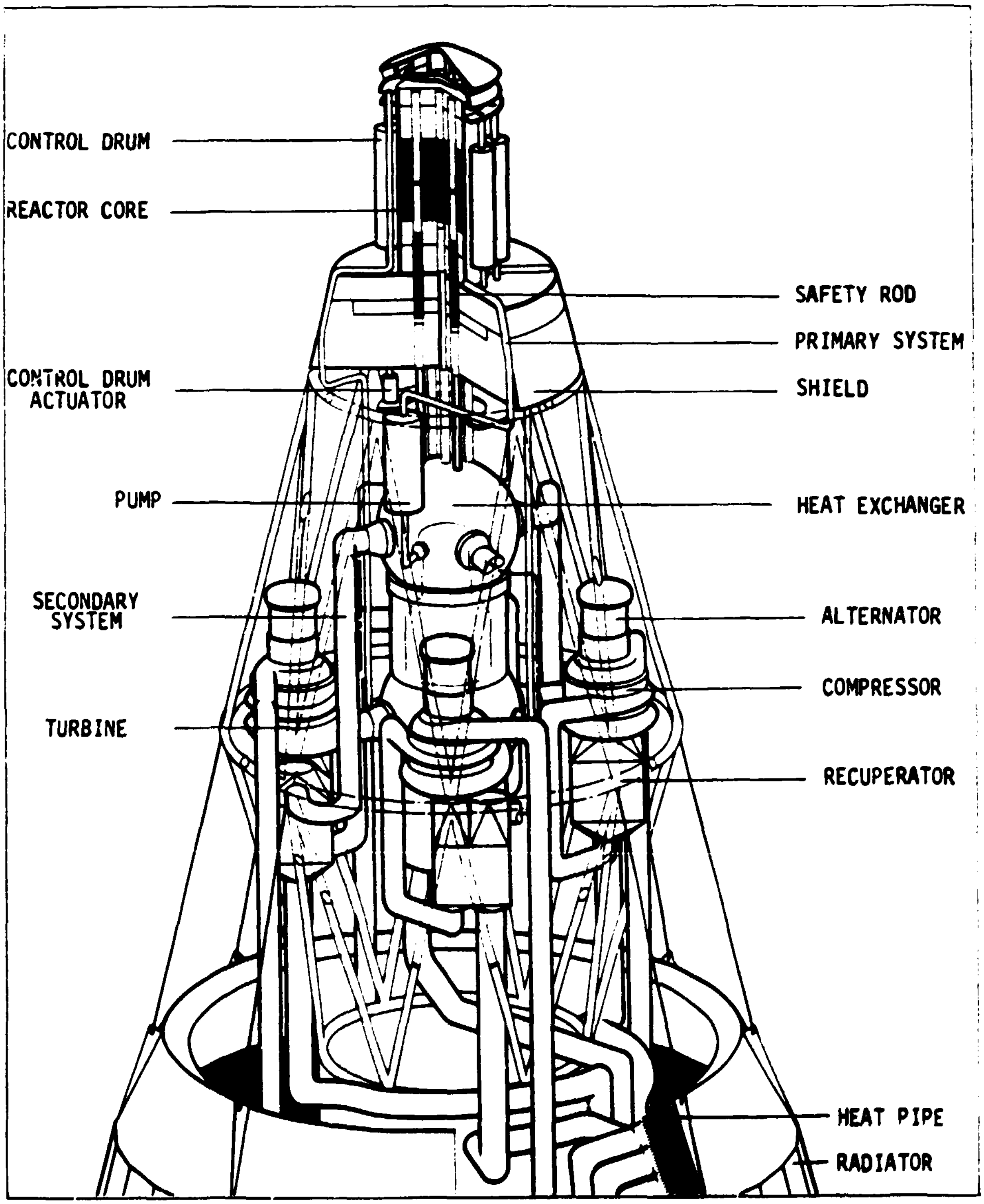

Figure 12 Layout of the 200 kWe ERATO Space Power System

ERATO

Nuclear space power reactor 200 kWe