BASIC FIGURES CONCERNING THE OPERATION OF A NUCLEAR REACTOR

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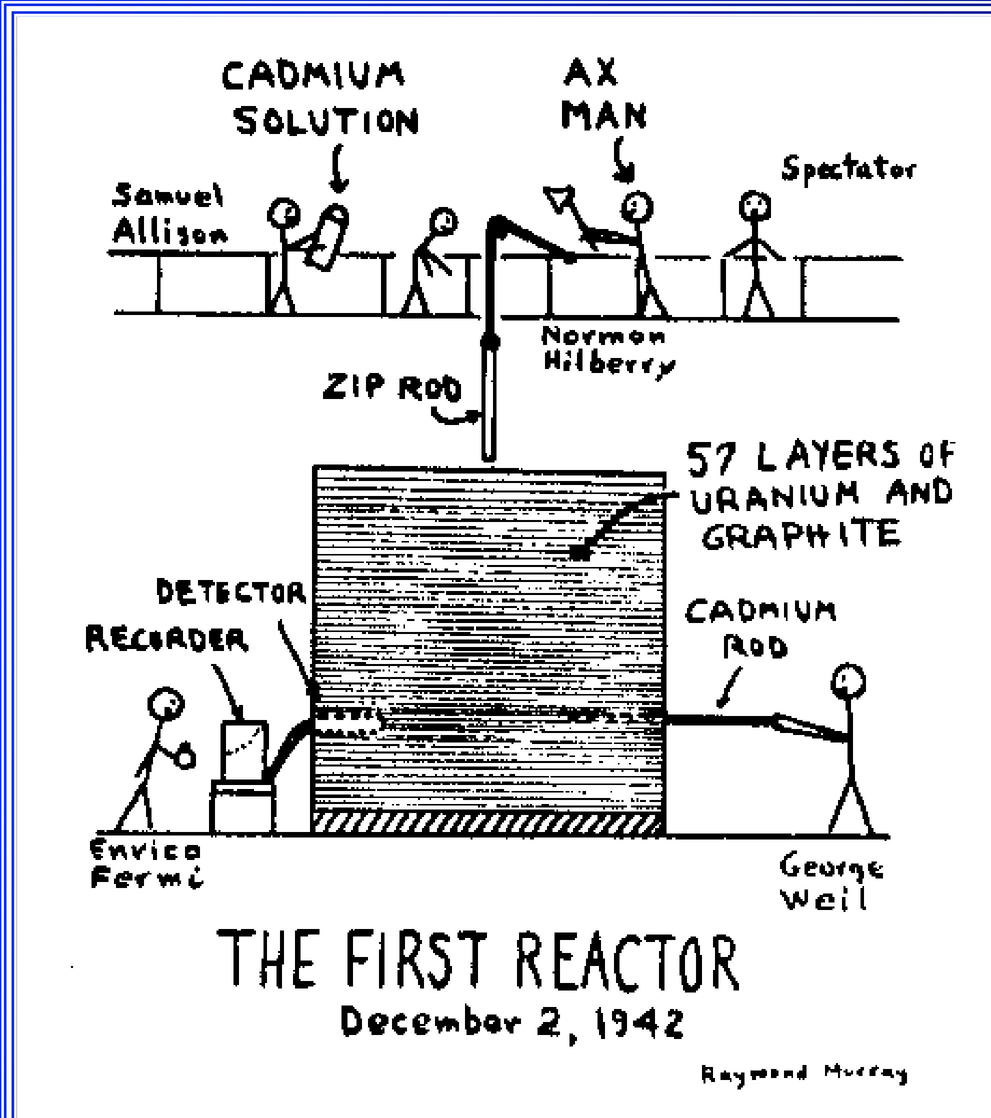
Enrico Fermi 1901-1954



Two types of nuclear fuels:

- **Fissile:** ^{233}U ^{235}U ^{239}Pu ^{241}Pu
- **Fertile:** ^{232}Th (□ ^{233}U) ^{238}U (□ ^{239}Pu)

Fraction of ^{235}U, in natural Uranium

$$\square_{\mathrm{nat}} = 0.7\% \; (= {}^{235}U/{}^{238}U)$$

Average energy of the emitted neutrons

is 2 MeV (fast neutrons);

slow neutrons are required

for efficient use of ^{235}U.

Experiments with metallic Uranium

show that the chain reaction starts if

$$\square = {}^{235}U / {}^{238}U = 8.2\%$$

However: when neutrons are thermalized

$T_n = 0.025$ eV

then $\sigma_{fission}(^{235}U) = 590 \times 10^{-24} cm^2$

while $\sigma_{reaction}(^{238}U) = 2.7 \times 10^{-24} cm^2$

Therefore:

$$\varepsilon_{nat}\sigma_{fission}(^{235}U) / (1-\varepsilon_{nat})\sigma_{reaction}(^{238}U) =$$

$$= 0.7 \times 590 / 99.3 \times 2.7 = 1.5$$

fraction of ***n*** interacting with ^{235}U =

$$= 1.5 / (1.5 + 1.0) = 0.6$$

Since in a fission 2 new ***n*** are produced (on the average), then the average number of new ***n*** per thermal ***n*** in natural U is

□ = 0.6 □ 2 = 1.2

If 17% of thermal ***n*** are lost (***n*** absorption or geometrical losses)

□ □ 0.83 = 1.2 □ 0.83 = 1

and the chain stops.

Thermalization

Needed: 2MeV $\rightarrow$ 0.025 eV (a reduction by $\approx 10^8$)

In elastic collisions with ^{A}M (moderator), fraction of energy lost is typically **1/(A+1)**, therefore the number N of collisions needed for thermalization:

$$\mathbf{N = 8/\log_{10}[(A+1)/A]}$$

A = 1 (hydrogen) captures too many ***n***

A = 2 (deuterium) is excellent

(also D_2O, Oxygen does not absorb too many ***n***)

A = 12 (Carbon) is excellent, provided it is in the form of very pure graphite

(Bothe, Heisenberg and the nazi reactor at Hagerloch)

Light water reactors (LWR) can be used provided U is enriched: not less than 2.5%.

Actually, in LWR

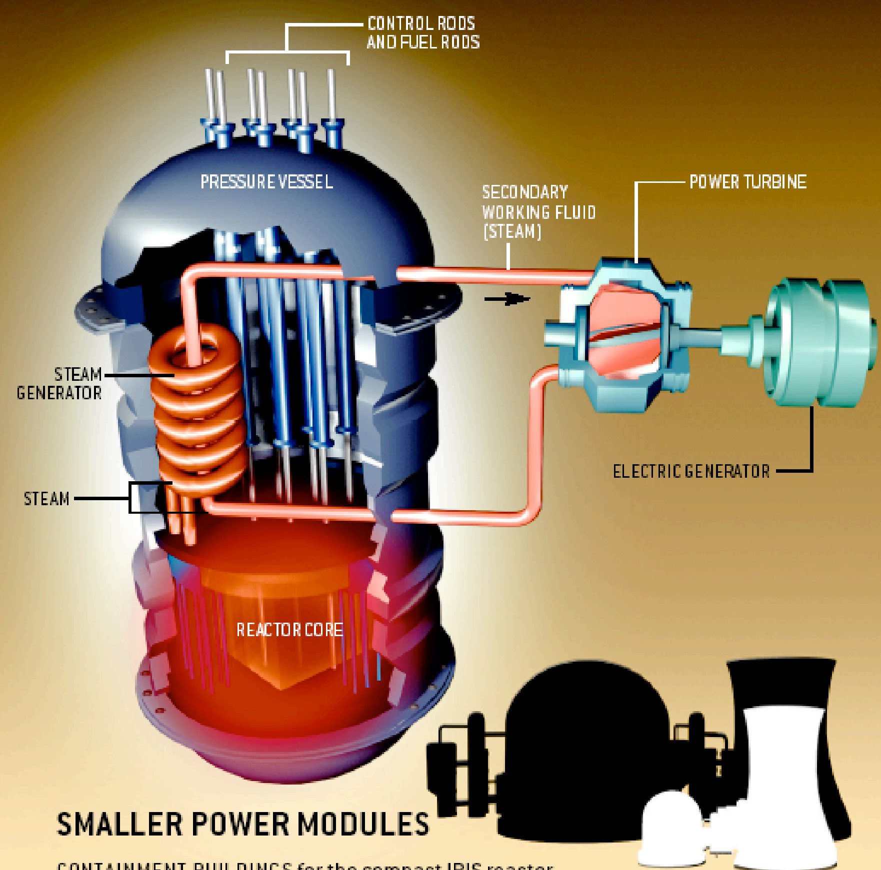
$$^{235}U/^{238}U = 3\%$$

As compared to graphite moderated, LWRs have some advantages: water is at a time moderator and refrigerator.

Enrichment costs because of chemical similarity of U isotopes.

WATER-COOLED NUCLEAR REACTOR

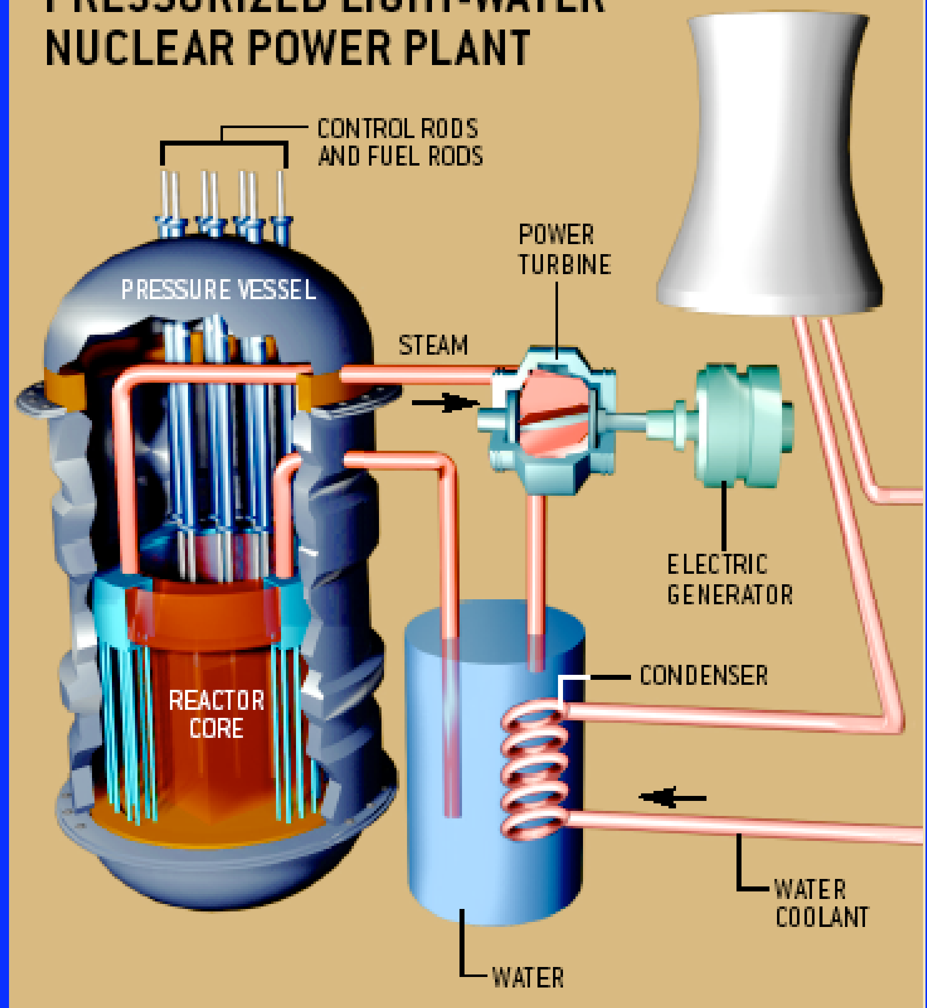
IRIS REACTOR DESIGN developed by Westinghouse Electric (*depicted in conceptual form*) is novel in that both the steam generator (heat exchanger) and the control rod actuator drives are enclosed within the thick steel pressure vessel.

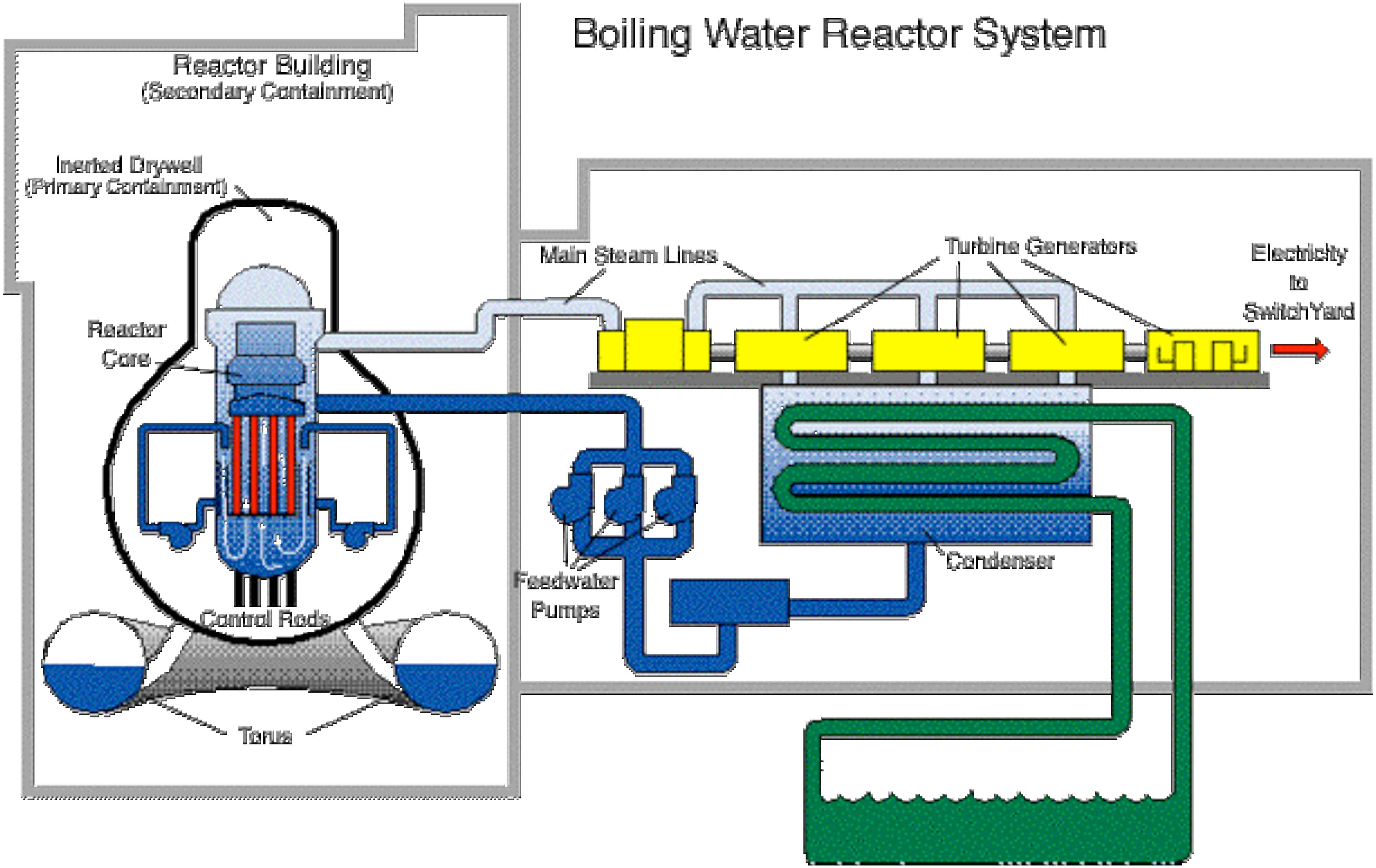


SMALLER POWER MODULES

CONTAINMENT BUILDINGS for the compact IRIS reactor can be reduced in size. The reactor's lower power output, ranging from 100 to 350 megawatts, can make these units more economical as well.

PRESSURIZED LIGHT-WATER NUCLEAR POWER PLANT




Boiling Water Reactor System
Reactor Building
(Secondary Containment)
Inerted Drywell
(Primary Containment)
Reactor
Core
Control Rods
Torus
Main Steam Lines
Turbine Generators
Electricity
to
Switch Yard
Feedwater
Pumps
Condenser

Graphite moderated reactors need no enriched U.

Fast ***n*** produce ^{238}**U** + ***n*** $\rightarrow$ ^{239}**U** undergoing
two β decays to ^{239}**Pu**.

^{239}**Pu** is fissile and:
(1) easily separated from U due to chemical difference
(2) produces more ***n*** per fission than ^{235}**U**

The less the fuel is enriched,
the more it produces ^{239}**Pu**.

Therefore:

military interest for graphite moderated reactors
(plutonium bombs or H-bomb triggers).

Pu is very convenient since bombs require
weapon-grade enrichment ≈ 90% because energy must
be rapidly liberated. Pu is more economic.

However:

$n + {}^{239}\mathrm{Pu}$ □ ${}^{240}\mathrm{Pu} +$ □

${}^{240}\mathrm{Pu}$ is not fissile by ***n***,
but makes spontaneous fissions
with emission of many ***n***.

If too much **${}^{240}\mathrm{Pu}$** in a bomb, it fizzles.

Because of **${}^{240}\mathrm{Pu}$** , fuel bars must be extracted with a periodicity of no more than 30 days to utilize Pu for bombs.

LWRs need 30 days just for replacing fuel: therefore refueling can be delayed up to every 1 or 2 years.

On the contrary, in graphite moderated (Hanford, Savannah River - USA, Cernobyl and many others - USSR)

the fuel can be replaced by extracting bars without stopping the plant; therefore they have no protection roof and are accessible from above.

LWRs are thermally stable.

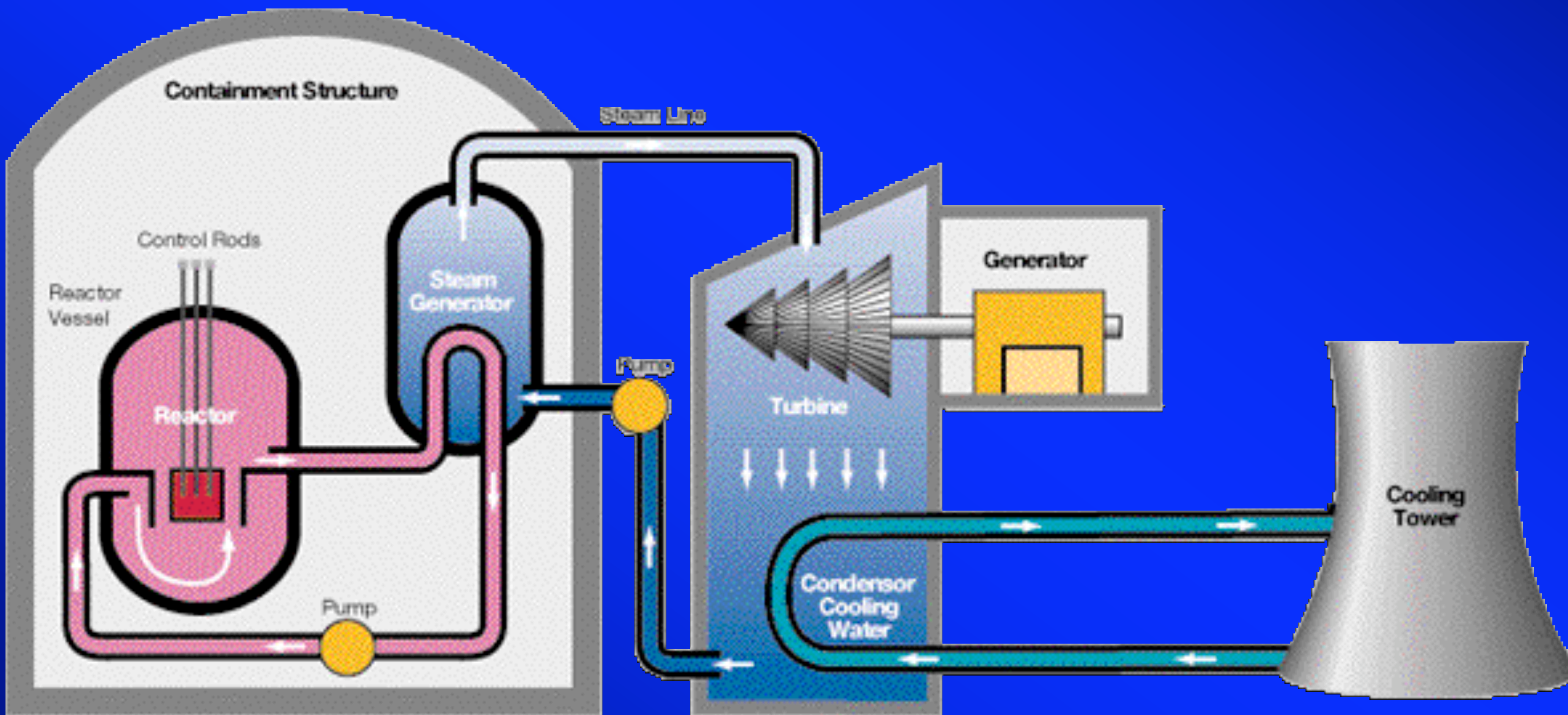
In case of LOCA

(Loss of Coolant Accident)

evaporation of water eliminates both the coolant and the moderator.

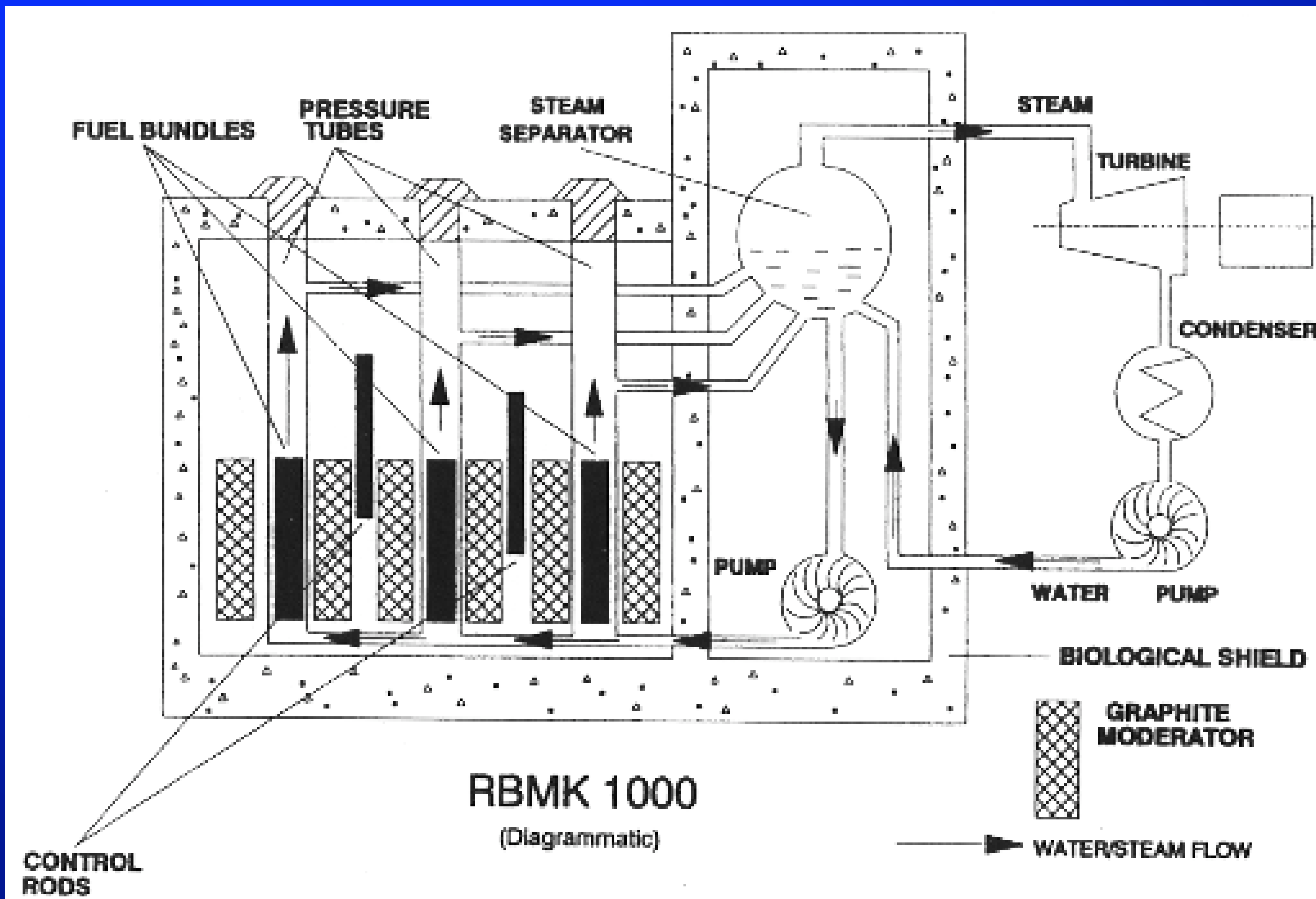
The chain stops. <u>However</u>: fuel melting problem. The residual activity of fission products (nuclear waste) amounts to ≈ 7% of the peak thermal power (≈ 3 times the peak electric power, typically 1000 MW).

This happened in 1979 at Three Mile Island, with no appreciable external consequences.



On the contrary, RBMK 1000 like Chernobyl, are thermally unstable: water is a “poison” because of *n* absorption. Therefore, LOCA is cathastrophic: the chain is less poisoned and accelerates, the produced power increases dramatically. Control must be “active”: control bars (cadmium) must intervene mechanically in the fuel mass. A too rapid temperature rise can impede the insertion by gravity of bars because of deformations of the seats.

Chernobyl


FUEL BUNDLES
PRESSURE TUBES
STEAM SEPARATOR
STEAM
TURBINE
CONDENSER
PUMP
WATER
PUMP
BIOLOGICAL SHIELD
GRAPHITE MODERATOR
WATER/STEAM FLOW
CONTROL RODS
RBMK 1000
(Diagrammatic)

Xenon interlude

Fission of **^{235}U** produces ^{135}I, which β decays into ^{135}Xe in 6.7 hours. Xe in turn β decays into ^{135}Cs in 9.2 hours. The capture cross section of ***n*** by ^{135}Xe is $(2.6 \times 10^{6}) \times 10^{-24}$ cm^2 !

The capture reaction is $\boldsymbol{n} + {}^{135}Xe \rightarrow {}^{136}Xe + \gamma$.

Capture is so frequent that ^{135}Xe not even decays into ^{135}Cs when the reactor works.

However, if reactor stops, ^{135}I decay continues to produce ^{135}Xe, which will not be destroyed any more and reaches such quantities as to empoison the fuel. For the same reason , any power reduction is difficult to control and reactors must work at maximum power continuously.

The sequence of Chernobyl accident on april 26, 1986 shows clearly that what happened was a thermal instability determined by the intention to avoid the Xe problem.

The control bars had no time to work: theydid not fall into their seats because these were endamaged by excessive temperature.

The plant exploded.

Comparison with bombs

A bomb differs from fuel on the time scale.
The difference is not the amount
of energy per kg (comparable)
but the time of release.

Since bomb fragments fly apart at
velocities of order 3000 m/s, energy
release stops in about 10□s
(the density of the explosive becomes
subcritical because the fragments separate)

Bibliography

- Bernard L. Cohen American Journal of Physics, 55 (1987) pg.1076

An excellent review of problems and comparisons

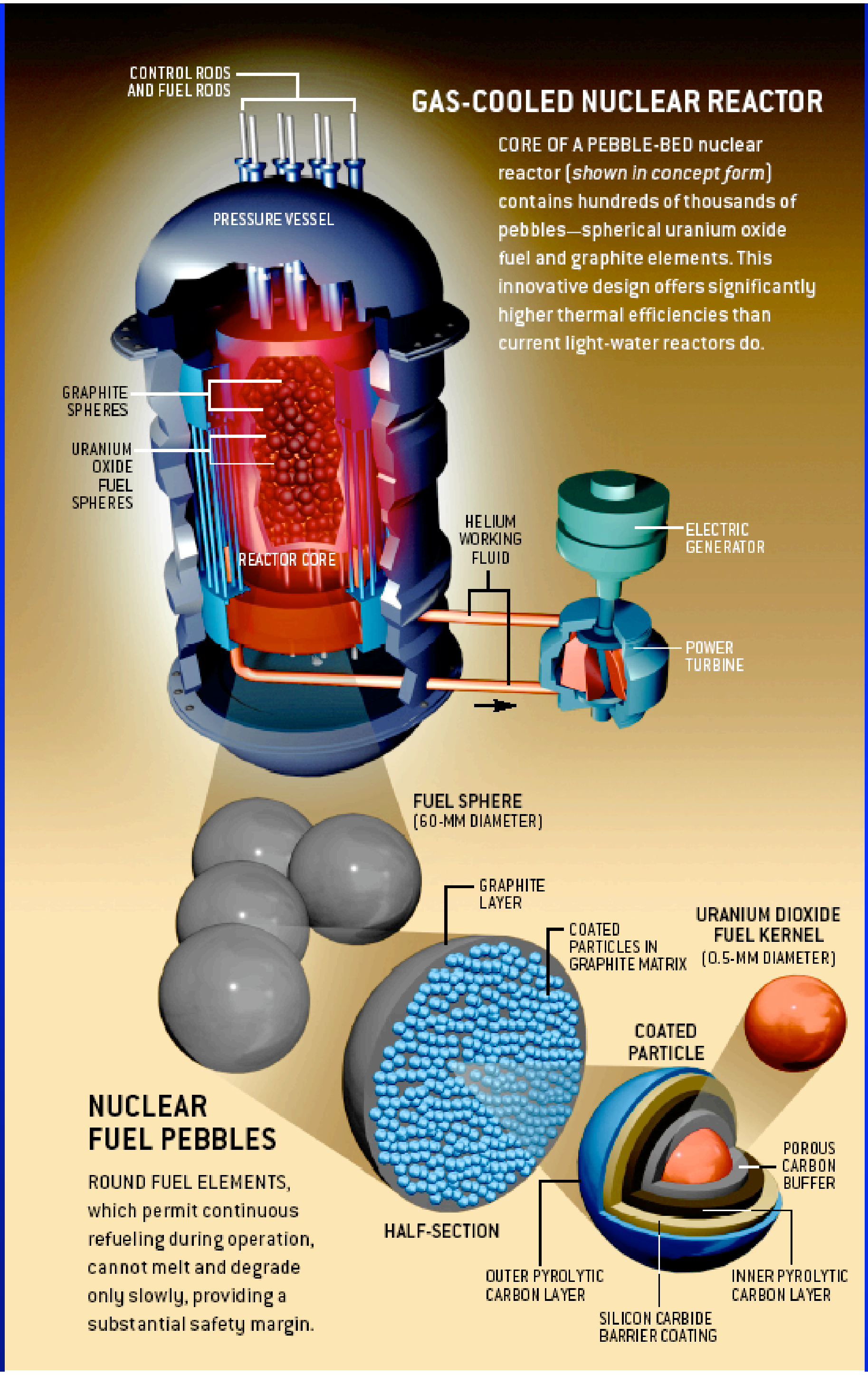
- IAEA, Board of Governors, Post accident review-meeting, Gov. 2268, 16/09/1986

Contains the detailed description of the accident at Chernobyl

- APS Study Group (R. Wilson ch.) Rev. Mod. Phys. 57 (1985) n°3, II part

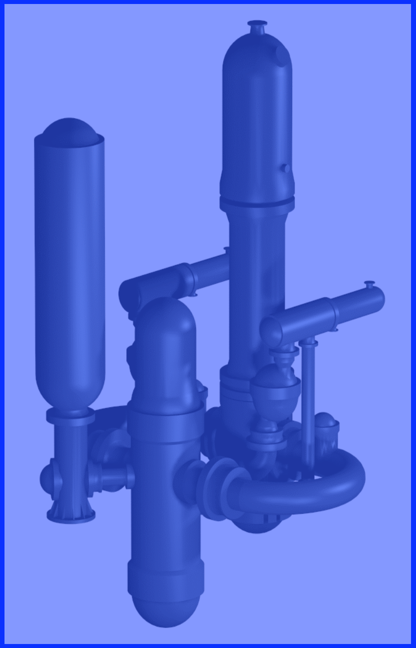
Contained warnings concerning safety on USSR reactors

- General view: shttp://www.nucleartourist.com/
- Safety:http://users.owt.com/smsrpm/nksafe/nineties.html#05



Gas-cooled nuclear reactor

Multipurpose Advanced Reactor inherently Safe



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MARS Main Design Objectives

- UNAFFECTABLE SAFETY
- REDUCED COSTS
- EASY DECOMMISSIONING
- MINIMUM WASTE PRODUCTION
- LOWEST DOSES

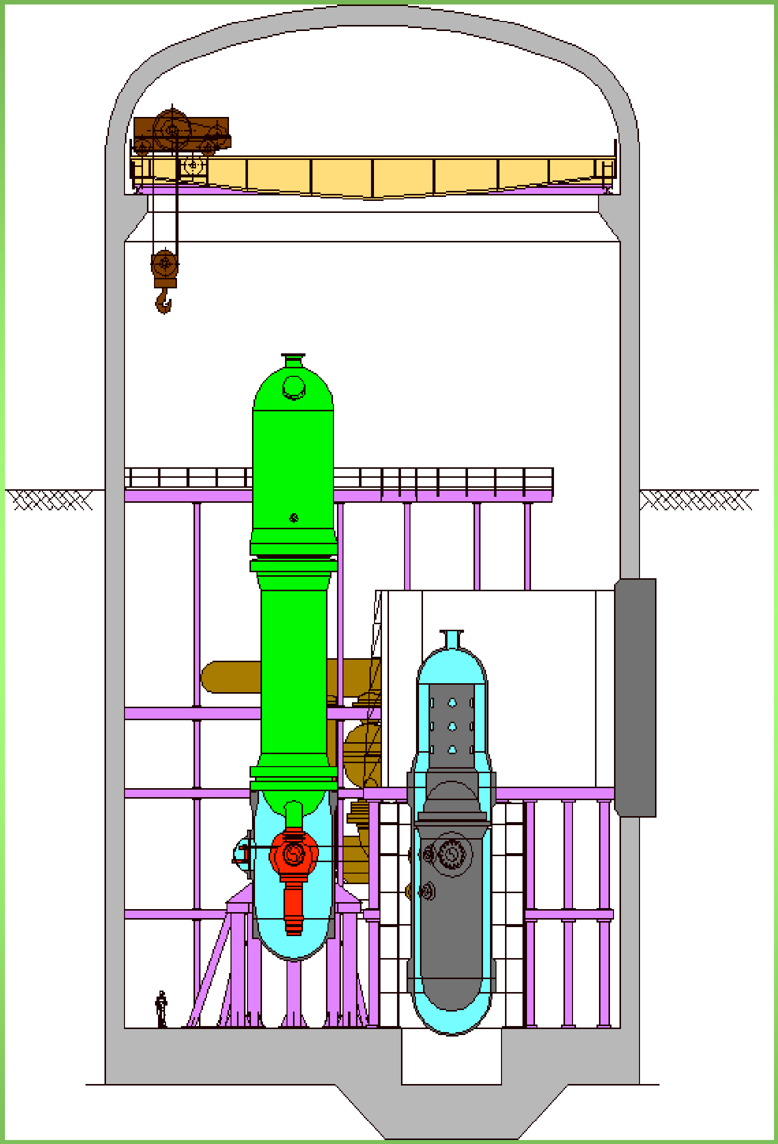
Main Design Criteria

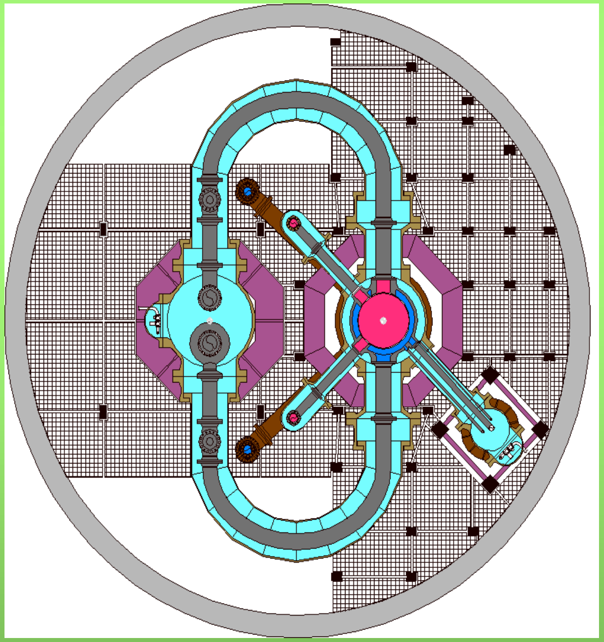
- MAXIMUM USE OF PROVEN TECHNOLOGY
- ADOPTION OF PASSIVE SOLUTIONS, AS FAR AS THE TWO CORNERSTONES OF NUCLEAR SAFETY ARE CONCERNED (*reactor shutdown and residual heat removal*)
- REMOVAL OF POSSIBLE PRIMARY-COOLANT BOUNDARY FAILURE
- PLANT SIMPLICITY
- REDUCED AND CERTAIN COSTS
- LOWEST RADIATION DOSES TO PERSONNEL
- LOW WASTE PRODUCTION

Main Characteristics

THE MARS REACTOR IS A PRESSURIZED LIGHT WATER REACTOR. THE PRIMARY COOLING SYSTEM INCLUDES ONE LOOP ONLY, WITH A VERTICAL-AXIS U-TUBE STEAM GENERATOR.

RATED POWER	600 MWt
OPERATING PRESSURE	75 bar
PRIMARY COOLANT FLOW-RATE	3227 kg/s
CORE INLET TEMPERATURE	214 °C
CORE OUTLET TEMPERATURE	254 °C
FUEL RODS ARRAY	17x17
FUEL ASSEMBLIES	89
TOTAL CONTROL ROD CLUSTERS	45





MARS containment building

Severe accidents

SEVERE ACCIDENTS IN THE MARS REACTOR ARE PHYSICALLY IMPOSSIBLE.

NEVERTHELESS, ACCIDENTAL SCENARIOS INCLUDING CORE MELTING HAVE BEEN TAKEN INTO CONSIDERATION AND THE IN-VESSEL CORIUM COOLABILITY HAS BEEN ANALYZED.

THE PRESENCE OF WATER IN THE PRESSURIZED CONTAINMENT ENVELOPING THE PRIMARY COOLANT BOUNDARY MAKES IT POSSIBLE TO ACHIEVE A SAFE IN-VESSEL CORIUM COOLING AND MAKES EVEN A SEVERE ACCIDENT COMPLETELY MANAGEABLE.

Project Development Status

- The nuclear design of the core and of the reactivity control systems has been completed.
- The design of the primary coolant system has been completed.
- The design of the passive-type emergency core cooling system has been completed.
- The mechanical design of the additional, passive-type scram system has been completed.
- The design of main NSS auxiliary systems has been completed.

Project Status (cont)

- **The mechanical design of advanced solutions proposed for traditional components has been completed.**
- **The design and verification of the reactor building and internal supporting structures have been completed.**
- **The analysis of produced wastes has been completed**
- **The HAZOP Analysis and the Probabilistic Safety Assessment of the plant have been completed.**
- **The Safety Analysis regarding all nuclear accidents has been completed.**
- **The cost analysis of energy produced has been completed**

Project Status (cont)

- The study of coupling of the NSS system to a co-generation scheme including desalination has been completed.
- The decommissioning program of the plant is going to be completed.
- Experimental activities have been performed to validate the main aspects of the design

A PRELIMINARY SAFETY ASSESSMENT REPORT HAS BEEN OFFICIALLY SUBMITTED TO THE ITALIAN NUCLAR SAFETY AUTHORITY

Annual solid wastes production (m^3)

Annual solid wastes production (m^3)

Type	Original production	Waste production after traditional conditioning	Waste production after advanced conditioning	Traditional PWRs with traditional conditioning (same power level)
resins	0.65	1.3[1]	0.05[4]	
filter cartridges	1.7	1.9[1]	0.32[5]	
compactable DAW	7.5	1.4[2]	0.45[5]	
non comp. DAW	0.65	0.7[1]	0.7[5]	
mixed wastes	0.12	0.13[1]	0.13[1]	
chemicals	1	0.5[3]	0.04[6]	
total	**11.62**	**5.93**	**1.69**	**20**

(1) cask filling
(2) low-pressure compacting and cask filling
(3) neutralization and cask filling
(4) incineration and cask filling
(5) high pressure compacting and cask filling
(6) drying and cask filling

Waste production

ADDITIONAL CONSIDERATIONS

• *About extensive use of passive safety*

In the past the nuclear plants' complexity increased very much, mainly as a consequence of the introduction of safety and redundant systems.

Nevertheless, the exasperation of the complexity of a plant system may not always be the best solution of the problem of the safety guarantee.

In fact, the complexity may cause the invalidation of two aspects representing the essence of the "inherent safety" of a plant: its reliability and simplicity.

- ***About plant lifetime***

MARS plant life time is of the order of one hundred years, so much longer than in traditional PWRs.

This is a consequence of the low neutron irradiation on the reactor vessel and of the low operating temperature which guarantees a high reliability of steam generator tubes (reactor vessel irradiation is one of the limiting factors for plant lifetime in nuclear power plants).

In addition, the non-traditional mechanical design allows the easy disassembling, and substitution if necessary, of all activated components, including reactor vessel and steam generator tube bundle.

- ***About MARS containment building***

The Mars reactor plant is equipped with a pressurized containment filled with pressurized water enveloping the whole primary loop (CPP – pressurized Containment for Primary loop Protection).

The plant is also equipped with a containment building to face external events in accordance with Italian and European regulations (aircraft impact, tornado, explosions, etc.).

The need to face these external events make the containment building able to withstand any internal pressurization, also in the incredible event of a complete destruction of the core coolant boundary.

• *About accidents made impossible in the MARS plant*

LOCAs are eliminated thanks to the adoption of the pressurized containment of the primary loop.

As a matter of fact, in the worst accidents, the origin of the deterioration of the core cooling is a consequence of ruptures in the primary-coolant pressure boundary.

ATWSs are eliminated thanks to the adoption of the additional passive-type scram system

LOFAs are eliminated thanks to the adoption of the Safety Core Cooling System.

• *About MARS capital cost*

In spite of the presence of the water-filled pressurized containment, that in the MARS concept is a relevant system, the plant investment cost is kept comparable with the one of the cheapest traditional nuclear plants, thanks to the huge plant simplification (drastic reduction of the number of systems and components), the reduction of components relevant to safety, the reduction of concrete volumes and the maximization of in-shop pre-fabrication.

The total direct investment cost, including contingencies, has been evaluated equal to about 1650 US$/kWe (referred to a 3-unit, 450 MWe station).

On the other hand, the plant simplification also allows for a reduction of operation and maintenance costs, so the cost of electric energy produced (kWh) has been evaluated to be less than 3.5 US¢ during the debt period, to drop drastically to less than 1.4 US ¢ for a long life period thereafter.

• *About required R&D activities*

MARS concept does not require substantial R&D activities.

All basic aspects have been deeply investigated both theoretically and experimentally.

Several experimental facilities have been realized to test behaviour and performance of the safety core cooling system.

Experimental activities to test behaviour and performance of additional passive-type scram system are going to be carried out.

• *About required R&D activities (Cont)*

All components use well-proven technology.

No accessibility is needed for underwater components; all submerged components are flanged: even if they do not require maintenance, if for any reason the access to them is needed, they may be removed and easily maintained in dry condition.

As far as the seismic response is concerned, a full seismic analysis of the whole primary loop and of the enveloping pressurized containment was performed, together with dynamic analyses concerning pressurization/depressurization of the enveloping containment, showing that stresses meet requirements of design code (ASME code) and that any mechanical interference between the two fluid system components is prevented.

• *Markets interested in MARS plants*

MARS plant seems to have many characteristics of interest for a market sector which includes small utilities and limited electricity networks (thanks to its limited size) as well as large utilities and extended networks (thanks to modularity).

On the other hand, the possibility of co-generation, allows to increase the overall plant efficiency and makes it appealing for a large number of applications.