38

STATE OF DEVELOPMENT OF GAS COOLED REACTORS IN THE UNION OF SOVIET SOCIALIST REPUBLICS

V.N. GREBENNIK, I.S. MOSEVITSKIJ
I.V. Kurchatov Institute of Atomic Energy,
Moscow

Yu. SUKHAREV
Experimental Machine Building Design Bureau,
Gorki

Union of Soviet Socialist Republics

In present-day conditions, taking into account consequences of the Chernobyl accident and growing negative attitude of population to nuclear power it becomes clear that its further progress is possible only at a cardinal increase in a nuclear and radiation safety and operational reliability of nuclear reactors and a guaranteed protection of a man and environment from radioactive influence.

With possible expanding of nuclear power application for industrial and domestic heat supply and for generation of process heat, which will require a maximum approaching of an energy source to a consumer, the problem of safety and reliability of a nuclear power plant increase even more. In these case, such operating conditions of NPP must be ensured during all period of its operation at which its effect on population and environment lies in permissible limits and at any accident, including hypothetical, no necessity to evacuate population occurs.

This will require a radical improvement of existing nuclear energy sources, an increase of safety of systems and equipment as well as the development of basically new technical solutions and the introduction of perspective reactor concepts.

In this connection in the USSR, the improvement of nuclear PWR performance takes place with respect to attaining an ultimate safety and increasing a reliability of systems and equipment.

Promising reactor concepts, including HTGR-based reactors, a mastering of which would be a significant stage in the development of nuclear power conforming to up-to-date requirements on safety, are also considered.

High HTGR qualities on the safety and ecological efficiency are determined by unique inherent safety properties peculiar to reactors of this type, as well as by the possibility to use passive protection means for the NPP.

Among the main internally inherent means are:

- negative temperature and power reactivity coefficients providing the reactor core stability against disturbances in the whole working temperature range;
- one phase and chemical inertia of a helium coolant;
- an impossibility of melting-down of the core in which a graphite with high sublimation temperature (3600°C) is used as a main structural material;
- resistance of the core to violations in heat removal owing to a high heat capacity of the core;
- a good compatibility of core structural materials with a primary circuit coolant in all possible temperature interval;
- a lack of appreciable reactivity margins for burnup due to accomplishing refueling of the spherical fuel elements during operation.

The low power density of the HTGR core and the above-mentioned properties make it possible to ensure a passive residual heat removal from a reactor at any emergency situations associated with a failure of a forced cooling-down and a loss of coolant.

The combination of inherent and passive safety features ensures the HTGR high self-protectiveness, i.e. such a quality when at an occurrence of emergency situation the population is not subject to a dangerous radiation exposure even without any active protection means and the operating personnel intervention.

The reactor plant (RP) self-protectiveness is characterized by a high operability during all period of operation, an ensuring a passive heat removal, a capability of reactor self-shutdown at emergency situations.

Transient temperature processes in HTGR caused by an emergency situation or failure in the operating conditions proceed very smoothly. At a high thermal inertia a behavior of the system is predictable and the personnel has a great deal of time to take measures to eliminate the emergency situation-

High safety features of HTGR have been demonstrated in FRG on AVR and THTR reactors and in USA on PICH-BOTTOM and FORT-SENT VREIN reactors, as well as at carrying out R&D in these and other countries of the world.

The concept of a few reactor plants with HTGR was being developed in the USSR over a wide range of powers and for various purposes.

Pilot plants with VGR-50 MW(el) and VG-400 MW(el) have been developed up to the stage of engineering designs.

At the present time the efforts are concentrated on the project of pilot-commercial reactor plant VGM (PCRP VGM) of a modular type with unit thermal power of 200-250 MW (Fig.1).

The installation is designed to solve main scientific and engineering problems of construction of high-temperature gas-cooled reactors, to try-out equipment components and to show advantages of the given type installations having an enhanced safety and capability to generate a high-potential heat. It should be a prototype for creation of commercial reactor plants for various power and process purposes.

The modular concept implies a multi-unit installation of high power - by integrating several reactors with a unit thermal power of 200-250 MW.

The VGM pilot-commercial plant with the core and the heat-exchange equipment placed in separate steel vessels was admitted to be optimal based on the analysis of various modular reactor variant parameters.

The reactor core is a pebble bed from spherical fuel elements of a 60-mm diameter, placed in a cylindrical cavity of the graphite reflector.

The primary cooling circuit providing a heat removal from an operating reactor, includes a main circulation gas flower, a high-temperature heat-exchanger and a steam generator, placed in one power vessel.

At emergency situations, including the circuit depressurization a residual heat removal from the core is accomplished by a surface cooling system (SCS) consisting of three independing channels based on the passive principle of action.

A scheduled and emergency reactor shutdowns are accomplished by rods dropping fruly and a reactivity compensation system based on inserting small diameter absorbing balls (KLAK systems) into side reflector channels.

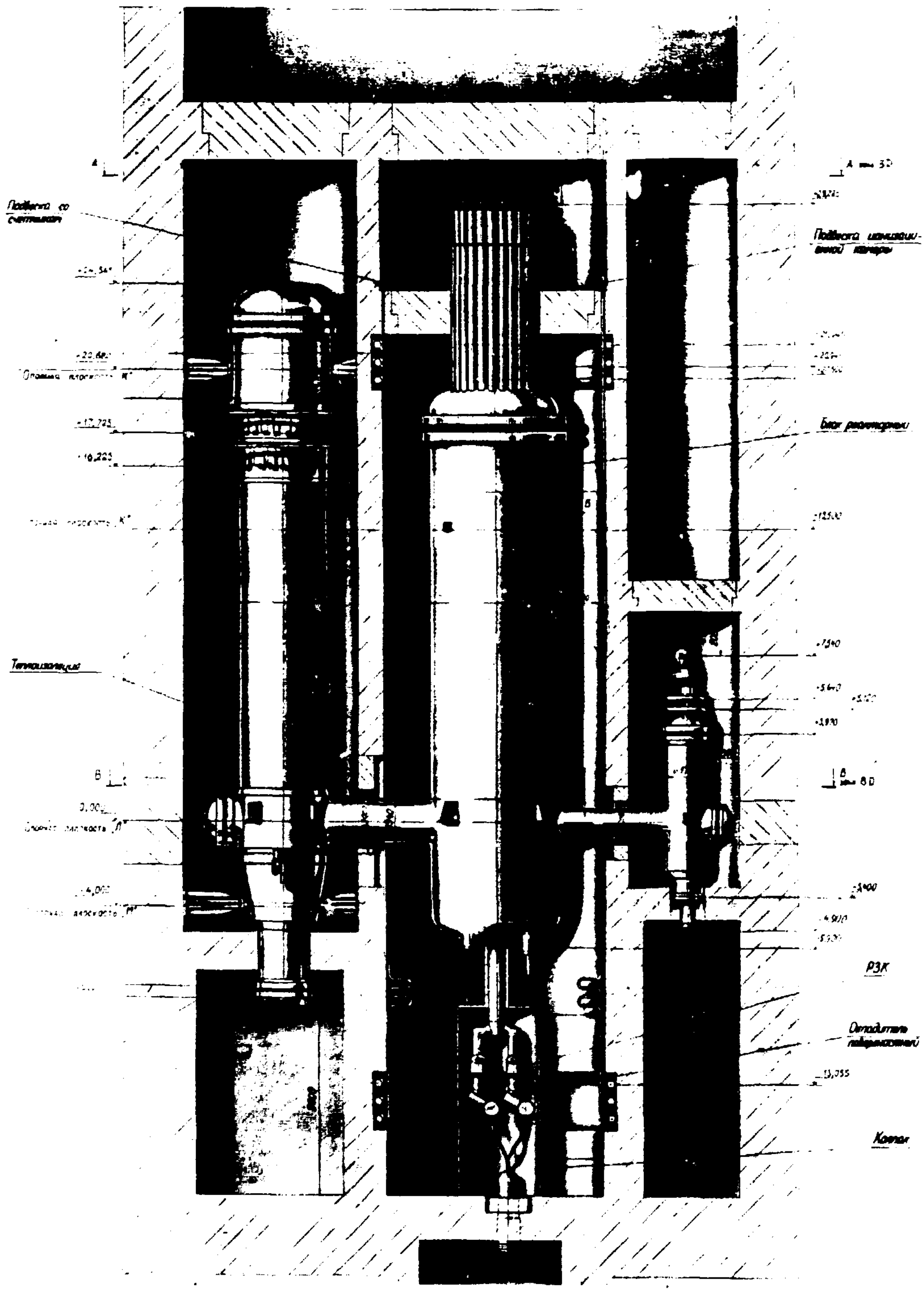

FIG. 1.

To protect RI and the main equipment from external exposures (shock waves, airplane crash, etc.) a protective containment is provided.

The use of SCS makes it possible in all design-basis accidents to ensure a fuel temperature of < 1600°C and a reactor vessel temperature of < 400°C.

Alongside with this for PCRP VGM an auxiliary heat removal loop is provided, with a power of 3% from nominal one which is ensured a long-duration removal of residual heat releases at a switching-off or failure of the main cooling circuit.

After confirming the reliability of the SCS and the operability of the vessel for multi-cyclic loads the auxiliary loop can be excluded from the commercial installations composition.

The VGM reactor main parameters are presented in Table I.

To substantiate the VGM design the R&D complex on main components of an equipment, structural materials and technology is carried out.

The study of movement of spherical elements was performed on reactor core models make in various scales. Coefficients of friction were determined and the thermophysical calculation of the reactor core was refined.

A complex of calculated-experimental and material testing studies was carried out to confirm operability of spherical fuel elements and the graphite reflector. Reactor tests have confirmed a high radiation resistance of fuel elements and the matrix graphite in all range of working fluences and temperatures.

Tests of more radiation-resistant marks of a stacked graphite satisfying the operating conditions in the VGM reactor are run.

TABLE I

Main Parameters for VGM Reactor

1	2
Reactor thermal power, MW	200
Primary circuit parameters:	
pressure, MPa	4.9
helium outlet temperature, °C	750(950-
Power density, MW/m^3	3.0
Core size, D/H m	3/9.4
Fuel element type and diameter, mm	spherical,60
Number of fuel elements in the core, pc	3.0×10^5
Fuel enrichment, %	8.0
Silicon mass in fuel element, g	10
Mean burnup depth, MW.day/t	8×10^4
Cycle length (average), eff.days	950
Number of absorbing rods, located in side reflector, pc	22
Number of channels of reactivity compensation spherical system (KLAK), pc	24
Ratio of fuel circulation through the core	10 - 15
Number of main cooling loops	1
Number of auxiliary cooling loops	1
Steam parameters:	
pressure, MPa	17.0
temperature, °C	540
Reactor vessel	steel

Tests of models and fragments of a steam-generator, a high-temperature intermediate heat - exchanger, gas blower elements, control rods, etc. are performed on helium facilities.

Certificate tests of structural materials loops for components of the equipment and units of the installation, operating at high-temperatures take place.

The carried out complex of works confirms a technical feasibility and expediency of creation of the new promising trend in nuclear power on the bais of modular gas-cooled reactors.

Opponents to the NP progress, in particular, to HTGR, point to their two main specific features which can be related to negative qualities. The first one - the graphite oxidation (burning) at a hypothetical accident with a reactor damage the consequence of which can be a rupture of fuel elements and a significant release of radioactivity. The second - the effect of water on the reactor reactivity, which can penetrate into the reactor core, the consequence of which can be a significant reactivity growth of the reactor in the worst hypothetical case. These are serious problems indeed which must be solved in developing the concept of ecologically safe reactor being stable not only to technical accidents, but also to sabotage and diversion.

As the performed development of VGM reactor indicate, the evolution of these accidents is predictable and does not result in catastrophic consequences.

Even at the most conservative assessment of the accident with the primary circuit depressurization the graphite oxidation in the reactor core proceeds extremely slowly and there are many tens of hours to search for and eliminate the accident.

The effective measure which reduces significantly and avoids completely the process of reactor core graphite oxidation at the accident with depressurization, can be the protection system based on a supply of inertial gas (nitrogen) to the reactor cavity during the accident.

The safety containment envisaged in the design serves as a protection from external exposures, prevents from radioactivity releases into environment, as well as restricts an intensive oxidation of the reactor core. A further way to exclude such accidents is the acceptance of the HTGR concept with protective barriers of fuel elements and on the reactor reflector, which ensure the fuel elements stability in an air medium at high temperatures in the case of any severe and hypothetical accidents. Such a barrier in the fuel cover or in the reflector surface layer should be completely or greatly tight.

The protective barrier not only protect the fuel element against oxidation and damage at its interaction with air or water, but also prevent the water absorption in the fuel element graphite, which excludes the possibility of an occurance of an emergency situation with the growth of reactivity.

There exists one more aspect increasing the safety, this is an underground location of NP, primarily its reactor compartment, which also ensures an additional protectiveness of NP from external actions.

Disadvantages of the modular reactor can be its comparatively high specific capital investments caused by a low power unit capacity, as well as by a low power density and an application of expensive high-temperature structural materials. However, this is the cost which must be paid for the NP high safety and for the possibility to use it for various power and process purposes, which will ensure force scarce kinds of organic fuel from the fuel-energy balance.

It should be noted that advantages of the modular-type plants manifest themselves at their quantity production and construction of multi-unit NP of high power, which will make it possible to reduce specific capital investments and to ensure the necessary competitiveness in the energy market.

Thus, one can believe that the suggested concept of ecologically pure modular HTGR facilitates to a great extent the problem of developing the reactor safe for population.

REFERENCES

1. N.N.Ponomarev-Stepnoi, I.S.Slesarev. The safety and efficiency of nuclear power- the basis in works at nuclear reactors of a new generation. Report at the IAEA General Conference, Sci. session, Vienna, September 21-25, 1987.
2. N.N.Ponomarev-Stepnoi, V.N.Grebennik. The state of development works on gas-cooled reactors in the USSR. Report at the 10-th International Conference on HTGR, San Diego, USA, September 19-20, 1988.
3. N.N.Ponomarev-Stepnoi, V.N.Grebennik, E.S.Glushkov, A.A.Khrulev, A.I.Kiryushin, V.V.Bulygin. Prospects of the development and the role of high-temperature,gas-cooled reactors (HTGR) in nuclear power. Report at the 11th International conference on HTGR, Dimitrovgrad, USSR, June 19-20, 1989.

STATUS OF THE FRENCH GCR PROGRAMMES

D. BASTIEN
Commissariat à l'énergie atomique,
Centre d'études nucléaires de Saclay,
Gif-sur-Yvette, France

1. PROGRAMMES

1.1 France has had no research and development programmes for high temperature gas-cooled reactors since 1979, when the decision was taken to end the studies carried out with the cooperation of the GAT on HTRs. The decision was taken for budgetary reasons, for it was no longer possible to develop three technologies at the same time. As it was decided in France to install PWRs in the short term and continue development of FBRs for the longer term, it was decided to abandon GCR technology. However, as the value of GCR technology is recognized in France, we have been continually monitoring developments in this field throughout the world. More particularly, a safety assessment is now being made in France of the modular concept now being developed in a number of countries. This activity forms part of a strategy of studying future reactors, including HTRs.

Furthermore, France still has a test loop in the SILOE research reactor (the *Comédie loop*), intended for studying fission product deposits. At the present time, a test programme for the DOE, of which MMES is in charge, is being prepared in collaboration with GA.

1.2 As concerns Magnox reactors, the CEA is carrying out, for EDF, operating and safety studies as well as the necessary studies for drawing up the dismantling files. In view of the progressive withdrawal of the Magnox reactors from service, the volume of this work is decreasing.

2. GAS-COOLED REACTOR FACILITIES

France has built 8 reactors on its own territory and one exported reactor at Vandellos in Spain. Only two reactors now remain in service.

2.1 Chinon A1 (70 MWe) was shut down in 1973 for financial reasons after ten years of operation. It has been transformed into a museum, representing an elegant solution to the