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GAS-COOLED REACTORS

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SUMMARY

At the previous ANS International Meeting in 1972, it was reported that six HTGR sales agreements had been announced in the U.S.A. Subsequently, these have been withdrawn because of pricing problems and changing market conditions. However, technology progams have continued and assessments indicate that the HTGR is competitive. A program to reestablish a commercial position is being conducted by industry and government. Also, an umbrella agreement has been proposed between the Federal Republic of Germany and the United States for information exchange and programmatic cooperation relative to gas-cooled reactors.

The high-temperature reactor HTR is very interesting due to its manifold possibilities of application for electricity production and nuclear process heat. In the long run there is the potential for nuclear heat to provide a large part of the required heat energy now produced mainly by imported oil and gas (e.g., 72 % of today's level in the FRG).

The AVR reactor in eight years of operation has demonstrated that the HTR technology with ball-shaped fuel elements can be realized. The radioactivity of solid and gaseous fission products in the primary circuit of the AVR have remained at low levels, and the mean outlet temperature of 950 °C has been demonstrated in continuous operation for more than 2 1/2 years. The U.S. Peach Bottom Reactor also operated well over its lifetime. Post-operation examination has confirmed that the circuit activities were low and that the second fuel charge was stable.

In the FRG the second pebble-bed HTR 300-MWe will go into operation in the year 1978.

The Fort.St. Vrain 330-MWe U.S. prototype HTGR is in the approach-to-power phase with the expectation of full power early next year.

The experience in the construction and operation of these units have clearly illustrated that a high safety standard can be realized. Experiments in the AVR have shown that even by switching off all safety devices (e.g., control rods, after-heat removal) no disturbances, especially no unpermissible high temperatures, result. From calculations and initial experiments it is to be expected that also for reactors of larger power the phenomena of core melting or core vaporization can be excluded. This can be ascribed to the increasing heat transport with increasing temperature in the reactor, to the large heat capacity, and the high melting points of the reactor materials. Due to the low ratio of power to heat capacity, effects of accidents are delayed by many hours. From experience with the release of fission products and their deposition it can be expected that the emission of radioactivity can be minimized even in very improbable accidents.

The breeder properties of the thorium cycle of the HTR reactor can be optimized by design calculations. Due to the continuous loading of the pebble-bed reactor, a conversion factor of more than 0.95 can be attained for a burn-up of 30,000 MWd/t. The required 233-U can be produced in the reactor by initial operation with 235-U and 232-Th. The uranium ore consumption of a reactor strategy with the pebble-bed HTR and an appropriate energy growth rate for Germany over a period of about 80 years is comparably not higher than the consumption of a strategy with LWR and LMFBR.

Until now, the capital cost of the HTR has been higher than the LWR although recent U.S. estimates indicate parity. Simplified components, especially the steam generators, are being designed to reduce cost.

Development of the fuel cycle technology is well along in both the U.S.A. and the FRG. A non-radioactive reprocessing pilot line has been constructed and prototypical refabrication equipment has been developed. A hot reprocessing pilot line is presently under construction. With fuel recycle, HTRs appear to have lower fuel cycle costs than LWRs, particularly as the cost of U_3O_8 rises.

The intensive development program for nuclear process heat should permit nuclear energy to substitute for imported oil and gas. The HTR with its high temperature capability is quite attractive for this purpose. The splitting of water is being tested in the laboratory and may be integrated into the development program. The aim advantages of nuclear energy for these processes are the saving of fossil raw materials and the better solution of environmental pollution problems.

Development for the gas-cooled fast breeder is being conducted primarily in the U.S.A., FRG, and Switzerland. Emphasis is being placed on overall design, consideration of program needs including prototype reactor tests and in-core heat transfer and fluid dynamic features. An electrically heated core simulation test facility and an irradiation test loop are being planned and built respectively in the U.S.A. and Europe to study core cooling under both normal and off-design conditions.

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Introduction

Experience to date with operation of high-temperature gas-cooled reactors has been quite favorable. Despite problems in completion of construction and startup, the three high-temperature gas-cooled reactor (HTGR) units have operated well. The Windscale Advanced Gas-Cooled Reactor (AGR) in the United Kingdom has had an excellent operating history, and initial operation of commercial AGRs shows them to be satisfactory. The latter reactors provide direct experience in scale-up from the Windscale experiment to fullscale commercial units. The Colorado Fort St. Vrain 330-MWe prototype helium-cooled HTGR is now in the approach-to-power phase while the 300-MWe Pebble Bed THTR prototype in the Federal Republic of Germany is scheduled for completion of construction by late 1978. The THTR will be the first Nuclear Power Plant which uses a Dry Cooling Tower. Fuel reprocessing and refabrication have been developed in the Laboratory and are now entering a pilot-plant scale development. Several commercial HTGR power station orders were placed in the U.S. prior to 1975 with similar plans for stations in the FRG. However, the combined effects of inflation, reduced electric power demand, regulatory uncertainties, and pricing problems led to cancellation of the 12 reactors which were in various stages of planning, design, and licensing.

The present emphasis in the U.S. is for an intensive new study of commercial possibilities for gas-cooled reactors. This is a cooperative effort between the Energy Research and Development Administration and private industry. Priority is placed on the steam-cycle HTGR, but the potential for future gas-turbine Brayton cycle and process heat applications are to be included. Also, the gas-cooled fast breeder role for the longer term is to be assessed. In Germany, a similar program is planned, with emphasis on the steam cycle. Moreover, there is in the FRG a considerably large project for the realization of the "Nuclear Process Heat", which for the time being is directed towards the gasification of coal.

In this paper we will discuss the operating experience and technology development, review the status of assessment and commercial studies, and look briefly at the potential for reactor more advanced than the steam-cycle HTGR.

Operating Experience

AVR

The first gas-cooled reactor in the FRG, the AVR, started operation 8 years ago[1]. It has a power of 50 MWth and 15 MWel and was operated at first at a helium outlet temperature of 850 °C. This outlet temperature was increased to 950 °C about 2 1/2 years ago. This shows that the pebble-bed reactor can be used for both electricity production as well as nuclear process heat. The reactor core consists of approx. 100,000 ball-shaped fuel elements having a diameter of 60 mm each. Upto now more than 2 million fuel elements have been circulated during operation in the reactor system without causing mechanical difficulties. Six different fuel element types have been tested. One group of fuel elements attained a burn-up of more than 150,000 MWd/t without any considerable damage. Corrosive and mechanical damages are constantly being investigated and it has been found that no serious problems occur.

The experiences regarding the occurrence of fission products have been satisfactory. They are an important precondition for the construction and operation of further larger high-temperature reactors. The free inventory of gaseous and solid fission products is so low that no serious danger can occur to the surroundings by the release of the coolant helium. This conclusion can be drawn by extrapolation also for large high-temperature reactors. Due to the low release rates of solid fission products, the contamination of the reactor circuit is small. The maintenance involves no problems.

When one considers that the fuel elements used today conform to the status 6 years ago, and that improvements have been attained in recent years, so it can be concluded that still better results could be obtained in later large plants. Improvements can be achieved, e.g. by getter-doping in the oxide core of the coated particles used, with the aim of retaining especially cesium and strontium even more efficiently in the oxide core.

The operation of the AVR should be continued for some years in order to demonstrate the high-temperature potential, especially for applications of nuclear process heat.

Peach Bottom Reactor

The U.S. Peach Bottom HTGR (40 MWe) operated on a planned program from 1966 to 1974. The first fuel loading was of an early design which led to fuel element cracking and release of some fission products to the reactor circuit. Even so, Core I operated with an overall plant availability of 58 % (removal of broken elements accounted for 72 % of the down time)[2]. The second core with a more modern coated particle fuel design operated satisfactorily throughout its rated lifetime of 900 effective full power days, with an overall plant availability factor of about 80 %. Further, operation of the various reactor systems such as the oil-lubricated circulators (and their seal and purge systems), control rod drives, and helium purification systems was very satisfactory for a prototype plant. Since permanent shutdown, an extensive post-operational test program has shown that activity levels in the reactor coolant circuit were quite low, as expected, and that otherwise the system is in good condition. Thus far, no evidence of deterioration of metal components has been found which would limit the lifetime of structural materials. Examination of the fuel elements confirmed the acceptable performance of the Biso-coated mixed carbide fuel particles [3]. Those portions of the fuel that operated at 1200°C or above showed significant redistribution of cesium by the purge stream to cooler parts of the fuel elements. In addition to Cs, significant levels of Co, Ag, Eu, and Sr have been observed in spine and sleeve graphite [4]. However, the nuclides which have migrated have still been strongly held by the graphite materials.

Dragon Project

The Dragon Project, which has a joint undertaking of member countries of the European Nuclear Energy Agency, was officially terminated on March 31, 1976, after 17 years of existence. The Dragon Reactor (20 MWth) was built to demonstrate the feasibility of graphite-moderated reactors cooled by helium and using ceramic fuel. The reactor was operated successfully from 1965 to 1975 to obtain extensive fuel and structure irradiation data and experience with the maintenance of helium circuits [5, 6].

In early years of operation the Dragon reactor was troubled with water-side corrosion problems which caused tube failures in the primary heat exchangers and required frequent boiler changes. A design modification to improve the homogeneity of flow distribution resolved this problem. Experience with operation of the helium circuits was favorable. Low helium leak rates and high reliability of the circulators and other components provided a good operating record. Radioactive contamination of the primary circuit was lower than might be expected from fuel irradiation experiments, some of which were carried to failure. Consequently, components in the primary circuit were easily maintained or replaced.

AGR

The advanced gas-cooled reactor (AGR) was introduced in the U.K. to take advantage of CO_2 technology gained from experience with the Magnox reactors. Adoption of enriched oxide fuel in stainless-steel cladding allows operation at temperatures about 200 ^oC higher than in Magnox reactors, with resultant better steam conditions, higher thermal efficiency, and reduced core size. The change to oxide fuel has also improved safety aspects of the reactor. The AGR, like the Magnox reactors, has the capability of on-load refuelling.

On the basis of experience with the earlier Magnox reactors, the AGRs were committed to construction with the technological backing only of work done on a 30-MW demonstration plant at Windscale. Little work had been done on the main plant components of the large reactors and difficulties were encountered which led to delays in construction up to 3 years.

There are ten AGRs, with combined output of 6000 MW, in operation or under construction at five sites in the U.K. The first two AGR units at Hinkley Point and Hunterston are now operating and the reactors at Hartlepool, Heysham, and Dungeness will follow in 2-3 years. The first units are limited to 80 % power for about one year when an assessment of the high temperature structural materials behaviour in CO_2 will be made to determine if the reactors can proceed to full power.

The design and operating experience with AGRs is valuable since many of the engineering features including concrete pressure vessels, ceramic fiber insulation, and large steel or concrete closures will be used in future gas-cooled reactors. The experience gained in operation gas circulators and some of the problems encountered (and solved) with respect to gas channeling, flow-induced and noise-induced vibrations, etc., should be a useful guide in design of HTRs and GCFRs [7].

Fort St. Vrain

The Fort St. Vrain Nuclear Generating Station achieved criticality in January 1974 and underwent initial startup tests. The plant was shut down for steam plant equipment modifications and for correction of problems with helium circulator Pelton wheels and control rod drive mechanisms. Plant operation was resumed and the plant reached 2 % power in April 1975 before it was shut down because of high temperatures in the control rod drive penetration cooling water system, caused by an unpredicted hot helium flow into the lower control rod drive assembly. This downtime was substantially extended by compliance with NRC requirements to resolve cable separation/segration problems, improve fire prevention, detection and suppression, and provide an alternate method of accomplishing plant colddown.

Operation was resumed in July 1976, and was in the process of rolling the turbine when purge flow to one of four helium circulator interspaces exceeded operating technical specification limits requiring a reactor shutdown. The high flow was determined to be due to leakage at the flange joint between the circulator and the penetration liner. The circulator was removed and a replacement circulator installed and the plant resumed its rise to power in mid September.

Technology Status

HTGR Fuel Recycle Development and Demonstration

An optimized commercial economy for HTGRs benefits from the recycle of HTGR fuel because of the relatively high cost for storage of spent fuel and the improved performance with the Th-^{233}U fuel cycle. Because HTGR fuels are graphite based, the flowsheet for reprocessing and refabrication of the fuel is different from that for LWRs. Although much of the technology has been developed, it has not been proven at a demonstration scale. Accordingly, programs have been organized in the U.S. and the FRG to develop processes for recycle of HTGR fuels. The emphasis of the U.S. program has been on the development and demonstration of prismatic fuel, whereas the FRG program is for recycle of pebble-bed-type fuel.

The program in the U.S. encompasses all parts of the fuel cycle operation, including head-end processing, refabrication, waste treatment prior to waste isolation, irradiation testing of recycle fuels, and development of demonstration concepts. The participants in the ERDA-sponsored U.S. program are Oak Ridge National Laboratory as the principal contractor, supported by General Atomic Company primarily in reprocessing and by Allied Chemical Corporation in waste pretreatment.

The development program for HTGR fuel recycle in the FRG has similar program elements to that of the U.S. However, the emphasis has been on the reprocessing operations; in particular, at KFA a pilot plant called JUPITER is being installed for engineering-scale work on the reprocessing flowsheet using irradiated pebble-bed-type fuel. Some work has been done at KFA on fuel refabrication, particularly particle preparation, and at HOBEG on design of devices for particle coating.

The programs in the U.S. and FRG are being evaluated and redirected consistent with other parts of the commercialization studies. The U.S. and the FRG are in the process of exploring methods of conducting a combinde HTGR fuel recycle program and supporting a joint demonstration facility in order to economize on resources.

HTGR Performance and Application Analysis

Either thorium and uranium fuel cycles can be employed in HTGRs. Application of the thorium fuel cycle rather than the uranium cycle permits more energy to be extracted from U_3O_8; further, the thorium fuel cycle permits more economic power generation than does the uranium cycle, particularly at higher U_3O_8 prices. The primary advantage of using the thorium fuel cycle is the higher conversion ratio which is obtained. The features of favourable fuel utilization, the high thermal efficiency obtainable, favourable design relative to safety considerations, flexibility with respect to reactor siting, and potential high temperature applications make the HTGR a very useful and attractive reactor system.

The nuclear performance of an HTGR is dependent on a number of economic and related conditions. With present estimated costs for fuel fabrication, fuel reprocessing and fuel refabrication, an HTGR with a conversion ratio of about 0.66 has about the same economic performance when spent fuel is stored as when it is recycled. As the price of U_3O_8 increases, it becomes more economic to recycle fuel than to store it; further, the conversion ratio of the most economic HTGR increases. At a U_3O_8 price of about $ 100/lb, the most economic conversion ratio is about 0.8.

Development of HTGRs can be justified on the basis of providing better uranium utilization, improved potential for long-term economics, and additional flexibility with regard to fuel recycle alternatives. Further, HTGRs operate effectively in a breeder reactor economy in which there is excess fissile fuel produced by the breeder reactors. Under such circumstances, product breeder fuel produced for HTGRs would preferably be ^{233}U. Breeders and HTGRs constitute an attractive combination because the relatively high fuel conversion and thermal efficiency of HTGRs permits the ratio of thermal to fast reactors in a breeder economy to be relatively high compared with that in an LWR/FBR economy.

Relative to plutonium use in HTGRs, Pu/Th fueling appears economically attractive relative to Pu/^{238}U fueling. Further, if low-cost Pu were available, high conversion ratio systems (CR $\sim$ 0.9) could be economically attractive in HTGRs. With low fuel recycle cost, HTGRs can also operate as near break-even-breeders. Calculations made at KFA Jülich show that for a mean burn-up of 30,000 MWd/t, a conversion factor of 0.95 could be attainable. Furthermore, our investigations in Jülich show that for a uranium price of more than 60 $/lb, the costs of a high converter are economically competitive. A conversion factor of 0.95 can contribute considerably to the saving of uranium ores. A study on the comparison of the uranium ore consumption for a LWR/LMBR-system and a thorium-high-converter-system shows that, for a time interval of 80 years, both these systems have a nearly equal uranium consumption. According to this study, the conclusion can be drawn, that the thorium cycle can be applied for a long-term period.

Advanced Gas-Cooled Reactors

In addition to the main-line steam-cycle HTGR, other designs for gas-cooled reactors are of importance. These are: (1) the gas-turbine direct (Brayton) cycle reactor (DCR or GTR); (2) process heat with utilization directly from the helium (VHTR); and (3) the gas-cooled fast reactor (GCFR). None of these has received a large effort for development or is approaching commercialization, but enough work has been done to indicate technical feasibility for each.

General Atomic and General Electric in the U.S. have conducted studies which indicate advantages for the DCR. The DCR couples well to dry-cooling towers where these are required. Also, the gas turbine is potentially a highly reliable machine. For these reasons interest has been indicated by utility groups in the U.S. However, the present level of effort is quite small. In the FRG, further development of the DCR will largely depend on the result of planned assessment studies, although a gas-fired helium turbine test loop now completed will be operated to obtain basic technology information.

Process Heat Applications [8]

The energy problem of many countries, so also the FRG, is limited not only to the production of electricity. The main problem is making the energy available for heat production. At present this part of the energy market is being covered mainly by oil and gas. Through this, there is an import-dependency in the FRG of more than 60 %. Due to known reasons, the task therefore arises of finding a substitution possibility for oil and gas. With this development aim, the project PNP (Prototyp Nukleare Prozeßwärme) was founded in the FRG, which is being carried out by the KFA Jülich together with a number of industrial firms. The aim of our project is to produce gas and liquid fuels by conversion of coal. The economy studies show that processes with nuclear heat have various advantages as compared to conventional processes. The amount of hydrocarbons produced is nearly twice as high based on the amount of coal employed. Electricity can be generated as a by-product, and the amount of CO_2 released to the environment is reduced. The plant costs for nuclear and conventional refining processes are about equal, so the application of nuclear heat yields cheaper products.

The fundamental precondition for the application of nuclear heat for such substitution processes is a helium outlet temperature of the reactor of 950 ^{o}C, which is attained in pebble-bed reactors with OTTO loading. The ball-shaped fuel elements enter the reactor at the top and then flow once through the reactor. Through this fuel distribution, the neutron flux, power and temperature distribution are such that for a helium temperature of 950 ^{o}C, the temperature of the coated particles reaches a maximum of only 1050 ^{o}C. Also a uniform burn-up of the fuel elements is achieved.

The first development aims of the application of nuclear process heat are the gasification of coal and lignite by steam and by hydrogen. For these processes, 3 small pilot plants were constructed in recent years, which demonstrate that a connection is possible between nuclear reactor heat and gasification processes. The heat transfer from the helium circuit in a fluidized bed for steam gasification occurs via an intermediate circuit with helium. The gasification with hydrogen is carried out in a reformer furnace in connection with hydrogenating coal gasification. The results abtained until now allow the design of a larger plant of approx. 750 MWth, which should be constructed in the next decade.

The HTR in connection with the reformer furnace can also be directly utilized for heat production and transfer, without the use of fossil fuels. In the reforming of methane by nuclear heat and steam, chemical heat is brought into a closed gas circuit. After cooling the gas mixture produced, which consists of carbon monoxide and hydrogen, it can be transported to the consumers. The utilization of the transported heat occurs through the synthesis of methane from the gas mixture, during which the applied reactor heat is released again. The methane so produced is led back to the nuclear plant in a closed circuit. The methanation process in which the reactor heat is released occurs at a temperature of appox. 500 to 600 ^{o}C, so that there is a broad spectrum of consumers e.g. for electricity production, for household heating and for process steam in industrial firms. A larger number of consumers situated at distances of upto 100 km can be supplied by electricity and heat by large nuclear plants. A first electrically-heated plant has been ordered and will go into operation as a demonstration plant in about 3 years.

The application of nuclear heat for the production of hydrogen from water has also made progress in recent years. An ideal solution for this process has, in my opinion, not been found yet.

The solution of the problem, of producing hydrogen with the help of nuclear heat, seems to be long-term of considerable importance. For many countries without fossil energy containing raw materials, it is the sole possibility of producing heat in sufficient amounts.

In the U.S., the Oak Ridge National Laboratory (ORNL), in cooperation with private industry, recently completed an assessment [9] of the technical and economic feasibility of the VHTR with particular attention being given to the potential for process heat applications. The study concluded that VHTRs have unique capabilities which make them attractive for use in synthetic fuels production, including hydrogen, and for generating intermediate load and peaking power. The costs for nuclear process heat are expected to be comparable to process heat costs from oil at $ 8-12/barrel, or from coal at $ 1-2 per million BTU. Evaluation of the HTR technology indicates that process temperatures up to 1400 - 1500 °F are achievable with near-term technology. Process temperatures in the range of 1600 °F to 2000 °F are potentially achievable but would require a large materials development program. From the safety standpoint, an isolation loop appears desirable, and possibly mandatory, for process heat applications. Potential obstacles, besides satisfactory resolution of safety questions, include materials limitations, adequate availability of the reactor which could dictate use of multiple units, and mismatch between the reactor heat capabilities and customer requirements.

Gas-Cooled Fast Breeder Reactor

The gas-cooled fast reactor has been studied quite extensively by several projects including the Gas Breeder Association, GfK, Karlsruhe, KFA Jülich, and General Atomic. Consistently the studies show good performance potential and a power cost advantage over the LMFBR. However, these studies have very large uncertainties because of the relatively small base of experience for most design features. The heat transfer and fluid dynamics of fuel pins with roughened surfaces, gas circulator performance, and the design of major high-temperature structural and shielding members inside the pressure vessel represent areas having relatively large technological uncertainty. Substantial programs are underway in Switzerland, FRG, and the U.S. to develop the technology of enhanced heat transfer surfaces. Some early shielding studies are in progress, but little work has been done on the gas circulators. Fortunately, some of the experience from HTGR and AGR circulator tests and commercial operation should be applicable. In fact, HTGR experience is relevant for many circuit components including the concrete pressure vessel and steam generators, and GCFR steam generator temperatures are lower than those used for HTGRs.

The fuel design and performance of GCFR fuel assemblies relates closely to that for the LMFBR, although there are two significant differences which exist between fuel pin requirements of the GCFR and LMFBR. Favourably, pin spacings are greater for the GCFR and thus more swelling from fast-neutron damage can be tolerated. At the same time, the high coolant pressure necessitates equalizing the pressure inside the fuel pins to that of the coolant circuit. Two irradiation tests of this feature have been conducted in single-pin irradiations, respectively to 50,000 and 100,000 MWd/MT burnup by ORNL. Although examination of the second test is not yet complete, the results are generally favourable. Further, irradiation testing of a vented bundle is being conducted by FRG, with the experiment being initiated at this time.

In the U.S. there exists a substantial interest by utility companies joined by three European utilities to support GCFR work at General Atomic. This program concentrates on the design and licensing of a 300-MWe GCFR demonstration plant.

The GCFR, because of is potentially good performance, represents a backup or future alternative to the LMFBR. Since existing GCR reactor system technology and LMFBR fuel experience can be utilized heavily, some savings can be expected in the overall development of a GCFR. Helium circulator tests, PCRV features, emergency cooling systems, and fuel pin pressure equalization would require extensive additional work before a demonstration reactor could be put into operation. If tests of these features proved satisfactory, this reactor concept offers interesting possibilities for very competitive performance.

Safety Research

The AIPA study on the safety of the HTR, which has been performed in the U.S., has already demonstrated the high safety potential of this reactor. In the FRG there will be also carried out a program for research on the safety on the HTR. Important results and aims of this program will be mentioned in the following.

Operational experience of the AVR, the DRAGON, and Peach Bottom reactor have shown, as mentioned previously, that the contamination of the primary circuit with gaseous and solid radioactive fission products is exceptionally low. It is therefore an important characteristic of high-temperature reactors, that for a loss of helium out of the primary circuit, only a small activity is to be expected in the containment and surrounding. Further safety experiments,assuming complete failure of all safety devices, were carried out in the AVR. The shut-down rods were withdrawn and fixed there and so made ineffective. Furthermore, the after-heat removal was interrupted after switching off the circulators, so that conditions were created which normally lead to the maximum thinkable accident. Under these conditions the reactor inherently stopped operation under full load conditions. As a result only a small increase of temperature in the reactor was to be observed.

The case of an accident of loss of coolant and failure of shut-down devices can be controlled in this reactor without any active measures. The results can be applied in a modified form also to large pebble-bed reactors. It was observed that the internal thermal conductivity increased with rise in temperature, so that in reactors of 3000 MWth also a large delay in temperature increase and a prevention of core melting or evaporization is to be expected. A power density of 5.5 MW/m^3 was assumed here. This power density is, on the one hand, sufficient to prevent core melting in a pebble-bed reactor; on the other hand, this power density value lies very close to the region of the most economic optimum.

With these safety characteristics, it is being attempted to attain an optimum safety concept, in which not only is the probability of occurrence of damage kept small, but also the maximum thinkable damage extent is reduced to such as is more acceptable for the public. The safety principle of the high-temperature reactor is based on the fact that both damage causes (1) emanation of helium, (2) release of fission products from the fuel elements, are largely decoupled in time. In the case of the biggest hypothetical accident (failure of after-heat removal, failure of shut-down systems), helium first emanates due to the increase in temperature, while the release of fission products occurs after a time delay of 3 to 4 hours. The transportation of the fission products, even in the case of simultaneous destruction of the primary circuit and the containment, can only occur by convection and diffusion. Herein lies the advantage of the high-temperature reactor, compared to present safety concepts, to reduce sufficiently and acceptably the maximum thinkable damage extent independently of the triggering cause.

An unpermissible increase in temperature after failure of all safety devices can be prevented in the pebble-bed reactor by the rapid removal of the fuel elements. By opening the by-passes, situated next to the normal removal devices, the fuel elements can be emptied into the fuel element container within approx. 1/2 hr and the whole system is thus brought to a safe condition. The rapid removal can occur either mechanically by hand or occur automatically with a high redundancy in case of exceeding a temperature limit of approx. 1200 °C. Thus there is a possibility of installing a reactor safety measure which prevents the exceeding of the temperature specifications of the fuel elements.

A safety problem of the high-temperature reactor which still has to be discussed is the penetration of air into the primary circuit after loss of helium. This danger is prevented by a burst-safe design of the primary circuit, so that only small and harmless amounts of air can penetrate into the primary circuit.

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