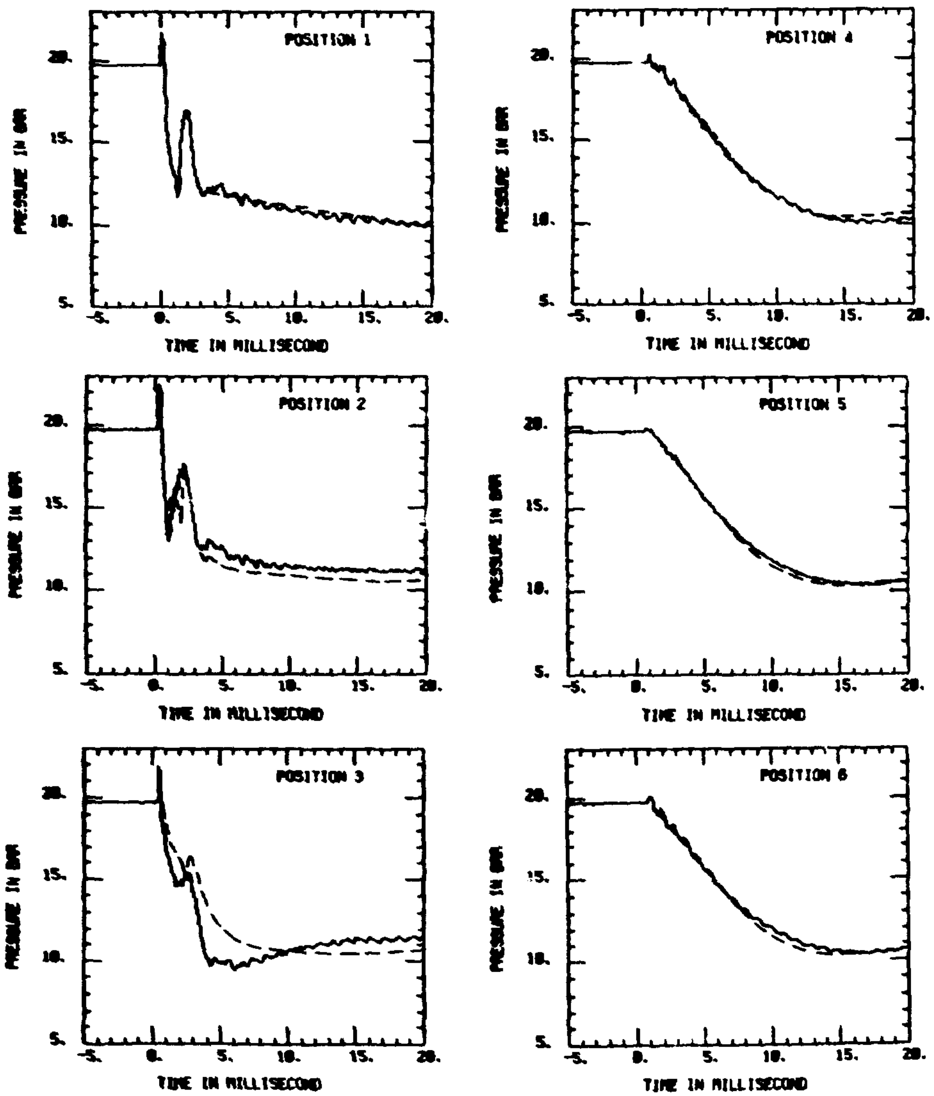
Fig. 6: Numerical results from the control volume method (CVM2)

——— Experiment

- - - CVM 2



THE HTR SAFETY CONCEPT DEMONSTRATED BY SELECTED EXAMPLES

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Abstract

The licensing experience gained in the Federal Republic of Germany is based on the licensing procedures for the THTR-300 and the HTR-1160. In the course of the licensing procedures for these reactors a safety concept for an HTR has been developed. This experience constitutes the basis for the design of future HTR's.

1. Licensing Experience in the Federal Republic of Germany

In the Federal Republic of Germany a great deal of experience has already been gained from the licensing procedure for High-temperatur Reactors. An experimental reactor, the AVR, has been in operation for more than 10 years and is demonstrating an extremely satisfactory operating and safety behaviour. A power reactor of 300 MW electrical power, the THTR 300, is presently under construction at Schmehausen (see Fig. 1).

The nuclear licenses for the erection of the various components of the THTR have already been granted in part or are being expected in 1980. The application for the operating licence has been filed and is expected to be granted in 1982.

In the years 1974 to 1976 a safety evaluation was carried out for an HTR with block-shaped fuel elements and an electrical power of 1160 MW. It was intended to construct this HTR 1160 next ot the THTR 300. The safety evaluation on concept and site was completed early in 1977 with a positive result. As usual, the safety evaluation report states various licensing conditions which remain to be met. This concernes essentially the accidents incurring water or steam ingress into the primary circuit.

The sequence of these accidents depends on the further failure assumptions which are different from those in the US. For this reason additional verifications are required in German licensing procedures.
After completion of the safety evaluation the HTR-1160 has not been persued in the Federal Republic of Germany.

The overall licensing practice and the experience gained from discussions with experts and authorities in the course of these projects furnish a solid basis for preparing a safety design concept for future HTR projects.

These are a steam-cycle plant, with an electrical output of about 900 MW, a process-heat plant,and a high-temperature reactor with helium turbine (HHT).

The fundamental safety requirements for the safety systems are greatly independent of the reactor concept, so that the available experience in safety design which mainly results from the experience gained with steam-cycle plants, can be transferred even to process-heat plants and HHT plants, although in these cases additional requirements must be taken into consideration such as the connection with the gas factory in the case of the process heat plant.

2. Safety requirements for high-temperatur reactors in the Federal Republic of Germany

In the course of the current licensing procedures the non-existance of HTR-specific safety criteria has proved to be a particular disadvantage. Since the BMI safety criteria, issued in 1974 by the Federal Ministry of the Interior, are based on the light-water reactor technology, their application to other nuclear power plant systems is problematic, because these criteria are based mainly on deterministic requirements, and less on the aspects of the protection of the environment. Thus they are not always appropriate to the safety characteristi of an HTR. The same applies to the regulations prepared by the Committee for Nuclear Technology (Kerntechnischer Ausschuß) in the Federal Republic of Germany.

The LWR safety criteria issued in 1974 and the subsequent application of these criteria to the THTR-300, whose construction had been started already in 1972, and further additional requirements resulted in subsequent modifications and important delay in the erection. It has been possible to carry out all the required changes without changing the basic design.

The safety concept remained however essentially unchanged.

In 1979 a draft of HTR safety criteria was compiled upon the order of the Federal Ministry of the Interior (BMI). These criteria were based on the current LWR-specific criteria containing mainly deterministic requirements. In part, however, also HTR-specific characteristics have been taken into consideration establishing a beginning to the compilation of the HTR-specific safety requirements.

These HTR-specific criteria, which will be taken as the basis of the design of the safety concept of future reactors, incorporate also prospective additional requirements as far as it seems reasonable (e.g. probabilistic requirements).

3. Safety Characteristics of an HTR

The safety concept of an HTR was based on the following HTR-specific features:

- System-inherent retention barriers for fission products

  The HTR possesses a system of barriers and delay ranges for fission products - coated particles, graphite matrix of fuel elements, prestressed concrete reactor vessel, and reactor containment. Especially the fuel particle coatings and the graphite matrix are resistant to high temperatures and retain their effectiveness up to extremely high temperatures.

- Unproblematic Coolant

  Helium, the coolant which is used in high-temperature reactors, is chemically neutral, it does not undergo any phase changes, and has a negligeably small reactivity influence.

- Negative temperature coefficient of reactivity

  The temperature coefficient of reactivity is negative over the total relevant temperature range.
  The reasons for the possibility to make use of this effect will be given below.
- Ratio of power density and heat capacity
  The ratio of power density and heat capacity which is decisive for the velocity of heating-up of the reactor core in the event of loss of cooling is lower in the high-temperature reactor than in the light water reactor by a factor of 10 - 15 due to the low power density (6 MW/m³ in the THTR) and the high heat capacity.
- Use of ceramic material

  The ceramic material used in the reactor core is characterized by an excellent resistance to high temperatures.
- Burst-safe reactor vessel
  The reactor pressure vessel is designed as a redundantly prestressed concrete reactor vessel into which all main components carrying primary gas are integrated.

The excellent fission product retention characteristics of the fuel elements ensure low contamination of the reactor coolant. (In the AVR the coolant gas activity is less than 100 Ci).
This results in:
- low exposure of the environment during normal operation and in the event of accidents
- low radiation exposure of personnel.

The experience gained in the course of the operation of HTRs (Peach Bottom, AVR, Fort St. Vrain) has shown, that the radiation exposure of the operating and maintenance personnel is lower than that of the light-water reactor by at least one order of magnitude.
The accumulated dose for the AVR plant personnel and the external personnel was measured to be 50 rem/a. The accumulated dose for the Fort St. Vrain power plant personnel is as low as approx. 3 rem/a.

In particular with regard to the coatings which are of decisive importance to the release of fission products the safety margin to failure is very large. The mean fuel element temperature in the THTR is 640°C, the design temperature is 1250 °C, complete failure of the coatings occurs only above 2400 °C. Diffusion of solid fission products starts, however, at about 1600 °C.

This chracteristic provides the possibility of actually making use of the negative temperature coefficient of reactivity acting as an inherent shut-down mechanism in the event of reduction or loss of core cooling, since in the HTR even a temperature equalization between fuel element center and surface (max. temperature difference approx. 200 °C) in case of interruption of heat removal cannot result in a destruction of the fuel element.
Due to the phase stability of the coolant, even in the event of "loss of coolant" there is always gaseous coolant available at a sufficient density. This means that loss of coolant is practically impossible. Only a depressurization of the primary circuit, resulting in a corresponding density change of the coolant occurs.

The extremely high heat capacity of the reactor internals with reference to the power density, results in an extremely slow temperature rise in the event of all reactor coolant accidents. In addition, part of the after-heat can be removed from the reactor core even in the event of loss of cooling, since the graphite internals have a high heat conductivity and heat capacity and the heat radiation becomes more effective in the reactor core at high temperatures. Also for this reason, temperature will rise very slowly in the reactor core. This means that plenty of time is available for countermeasures even in the event of hypothetical accidents.

4. The HTR Safety Systems

The safety systems of an HTR are mainly designed according to deterministic aspects such as required according to the safety criteria and practiced in the licensing procedure. This means that the safety systems are designed so as to control all assumed accidents taking into consideration a "single failure" and a "repair case" even in the event of failure of the first initiation of the plant protection system for the detection of accidents.

In the following the safety systems are briefly described with reference to examples:

Reactor pressure vessel

The reactor pressure vessel of all HTRs, except for the small experimental reactors such as the AVR in Jülich, are constructed from prestressed concrete.
Such prestressed concrete pressure vessels are considered as burst-safe, which has been confirmed in experts' opinions for the THTR 300 and the HTR-1160.

Shutdown systems

The design of the shutdown system for pebble bed reactors can be based on the experience gained with the THTR 300. In the THTR, the first shutdown system consists of 36 absorber rods in the reflector. For larger reactors with reactor powers higher than 300 MWe the effectiveness of the reflector rods is, however, not sufficient so that the reflector rods are backed up by absorber rods to be directly inserted into the pebble bed.

In the THTR the second shutdown system consists of 42 absorber rods which are directly inserted into the pebble bed. In larger reactors the incore rods also assume the tasks of a second shutdown system. For reasons of diversity, additional shutdown possibilities for future plants are being investigated such as injection of neutron-absorbing gases into the primary circuit or introduction of small absorber spheres into the pebble bed core. These additional shutdown possiblities are being investigated especially for large reactors in which scram cannot be effected by the reflector rods only.

A reactor shutdown resulting exclusively from the negative temperature coefficient acting as an inherent shutdown mechanism has not been used up to now instead of hardware measures. This characteristic is, however, of extreme importance to the control of hypothetically assumed accidents, thus contributing essentially to the reduction of the residual risk.

The favorable accident behaviour has been confirmed in the AVR by repeated "rod jam tests". In this demonstration test the coolant circulation was suddenly interrupted at full power and the insertion of all the four shutdown rods was prevented. Due to the negative temperature coefficient, power production was immediately reduced to residual heat level. The generation of xenon kept the reactor subcritical for about one day; then power balanced out in the range of a few kilowatts, according to the heat transfer to the internals surrounding the reactor core.

Afterheat removal system

In principle, afterheat removal in high temperature reactors can be effected by the main cooling system during all operating conditions and accidents. In the AVR, THTR-300, and Fort St.Vrain power plants this afterheat removal concept is being used.

In the THTR-300 an additional water injection into the 6 steam generators is provided in addition to the purely operational systems. For this purpose 3 steam generators each are combined in an emergency cooling circuit. In the event of grid failure electric energy is supplied by an emergency diesel for each circuit. If the reactor is under pressurized conditions, one of the six steam generators is sufficient for afterheat removal. The THTR-300 is the first reactor for which a risk evaluation was claimed in the course of a licensing procedure in the Federal Republic of Germany in connection with the licensing of the afterheat removal system. In addition to the verification of the required reliability of the afterheat removal system it could be shown in this connection that also in the event of failure of the complete afterheat removal there are approximately three hours available for initiating counter measures already planned in the afterheat removal system.

The HTR-1160 reactor project was the first project in which in addition to the main cooling system a further auxiliary cooling system was introduced, consisting of four afterheat removal loops (see Fig. 2). This afterheat removal concept obtained a positive evaluation in the course of the assessment of the HTR-1160 concept.

At that time, the reasons for adopting an auxiliary cooling system, independent of the main cooling system were

> that the main cooling circuit must not be designed to be fully functionable in the event of the maximum depressurization accident.
> (lower extent of verification, lower costs)

- the main cooling circuit with all its required auxiliary systems was not to be designed as a safety system, it was designed according to purely operational requirements.

- The auxiliary cooling system, which is independent of the main cooling system, represents a second afterheat removal system. This results in an increased safety potential.

The same concept is taken as a basis for the plants which are currently being designed, such as the steam cycle plant with a pebble bed reactor.

For a pressurized reactor one of the four afterheat removal loops with a maximum efficiency of approx. 100 MW is sufficient for the removal of afterheat; with the reactor depressurized, two of these loops are required.

A further increase of the reliability of afterheat removal will be possible by a separate water injection into the steam generators of the main cooling system. This would result in a further reduction of the residual risk, which is very low anyway.

5. Accident behaviour

In the licensing procedure practiced in the Federal Republic of Germany it is assumed that in principle all components can fail, the failure model depending on the design and the stresses to which the component is exposed. In addition, in the event of an accident consequental damages must be assumed to occur on the components which are not designed against the accident stresses with sufficient safety margins or which cannot be exposed to inservice inspections to a sufficient extent.

The design of the safety systems is mainly determined by the accidents to be assumed. In high-temperature reactors these are especially:

a) Depressurization accidents
b) Water ingress accidents

Depressurization of the primary circuit is not as serious an accident in an HTR as in a LWR in particular, with regard to the activity released. Special afterheat removal systems are not required for controlling this accident. Neither are there any consequential damages to be expected to occur on the fuel elements. Thus the activity release is relatively low.

If considerations for the THTR-300 are based on the design basis activity of 35 000 Ci, the loss of coolant accident will result in a whole body dose of about 50 mrem. This applies to an event, where the activity is immediately released over the stack without delay or filtering. In the hypothetical event of ground release the maximum permissible load in the Federal Republic of Germany of 5 rem would not be exceeded. Thus it has been possible to construct the THTR without the containment usually required for LWRs.

Future HTRs will be designed with a containment, e.g. to improve the possibility of controlling external effects. Fig. 3 shows the history of the maximum fuel temperature during a depressurization accident in the HTR-1160. This figure shows that there is a large safety margin to the failure limits of the fuel elements.
Apart from the depressurization accident, water ingress into the primary system due to steam generator damage is a decisive factor for the design of the afterheat removal systems. Therefore it is achieved by a high quality standard of manufacture and in-service inspections that major steam generator damages can be excluded.

For limiting undue corrosion of the fuel elements and the ceramic reactor internals the water ingress into the core in the event of a steam generator damage will be limited by dumping the defective steam generator and by reducing the graphite temperatures to levels below approx. 700 °C by a respective design of the afterheat removal system.

The effectiveness of the inherent safety characteristics in the event of a hypothetical accident has been analyzed by General Atomic in the course of risk assessment studies on the HTR 1160. (AIPA Accident Initiation and Progression Analysis). This study has since been transferred to German conditions by the Institute for Nuclear Safety Research in the Nuclear Research Center of Jülich.

In addition to the extent of damage, also the probabilities of occurrance and the risks of the various accidents have been determined. Among others, also the risk of hypothetical accidents with total failure of safety installations has been determined.

These analyses have shown that the risk of high-temperature reactors is comparatively low and clearly below the risk of other reactor lines.

AVR

START OF CONSTRUCTION	1961
COMMISSIONING	1967

THTR-300

START OF CONSTRUCTION	1972
APPLICATION FOR OPERATING LICENCE	1980
PROSPECTIVE COMMISSIONING	1983

HTR-1160

COMPLETION OF LICENSING PROCEDURE FOR CONCEPT AND SITE	1977

FIG.1: HTR IN THE FRG

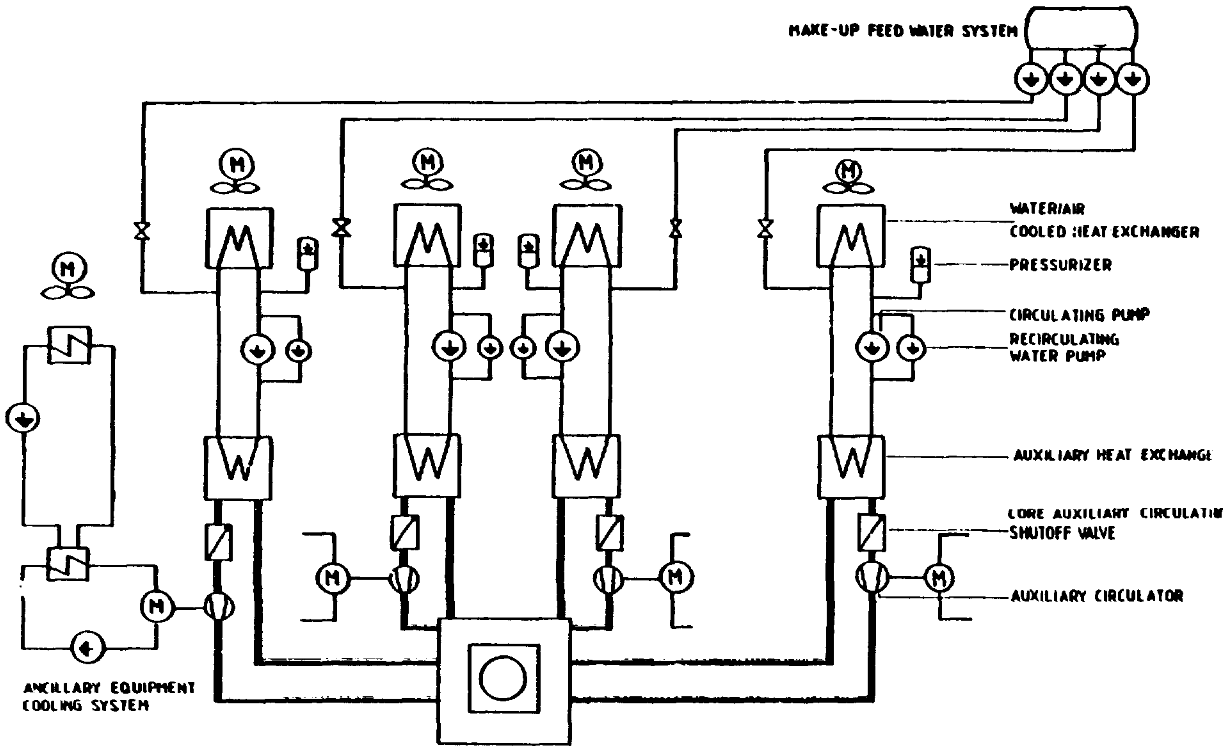

FIG 2 : CORE AUXILIARY COOLING SYSTEM FOR AFTERHEAT REMOVAL

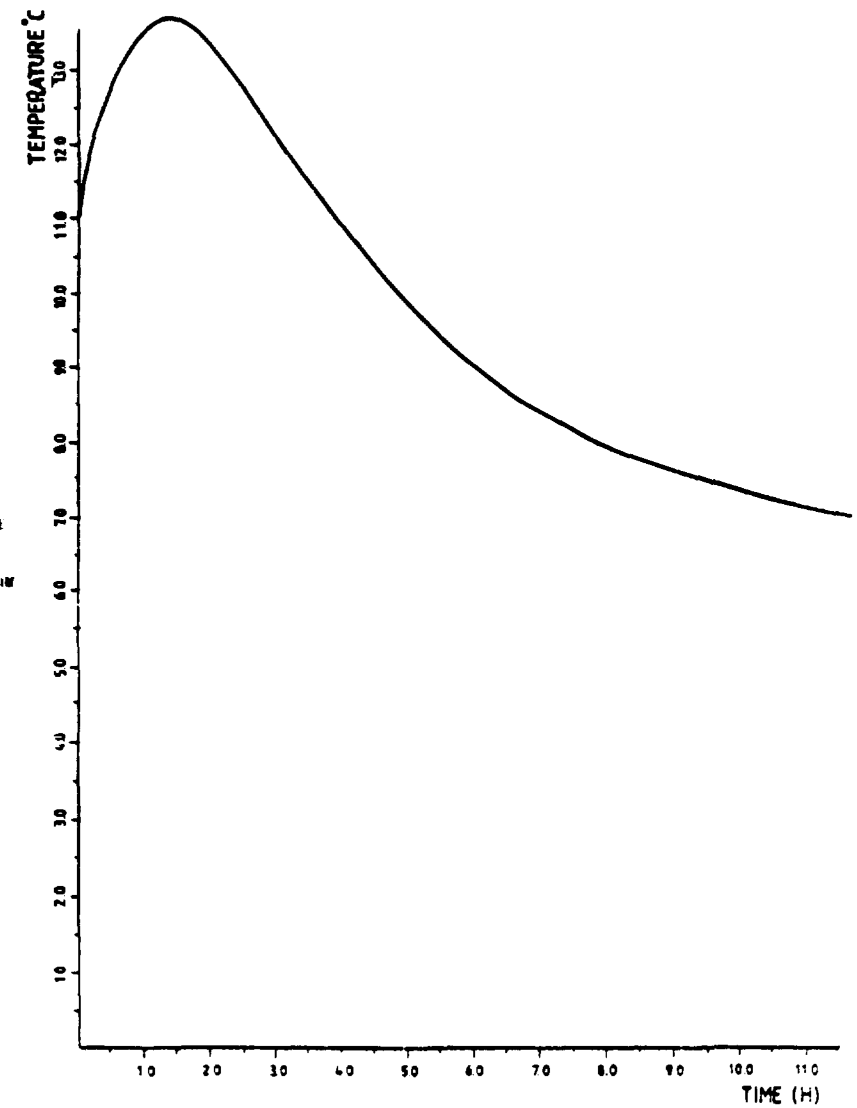

FIG.3 MAX. FUEL TEMPERATURE DURING AFTERHEAT REMOVAL IN THE EVENT OF A DEPRESSURIZATION ACCIDENT (2 LOOPS)