SUMMARY OF THE IAEA SPECIALISTS MEETING ON GAS-COOLED REACTOR SAFETY AND LICENSING ASPECTS

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Editor's Note: Following a recommendation of the International Atomic Energy Agency International Working Group on Gas-Cooled Reactors (IWGGCR), a Specialists Meeting on Gas-Cooled Reactor Safety and Licensing Aspects was held in Lausanne, Switzerland, September 1-3, 1980.

The gas-cooled reactor programs considered by the IWGGCR include carbon-dioxide-cooled thermal reactors, helium-cooled thermal high temperature reactors for electricity generation and heat production, and gas-cooled fast reactors.

This meeting had 52 participants from 10 countries presenting 28 papers. The three-day meeting included three sessions on the subject of safety and licensing aspects, including descriptions of national programs, experimental and theoretical results, and operational experience of gas-cooled reactors. A list of the papers presented at the meeting appears in the Appendix.

SUMMARY OF SESSION I: SAFETY AND LICENSING EXPERIENCE OF GAS-COOLED REACTORS (GCRs)

Contributions were made concerning the operating experience of the Fort St. Vrain (FSV) high-temperature gas-cooled reactor (HTGR) Power Plant in the U.S., the experimental power station Arbeitsgemeinschaft Versuchs Reaktor (AVR) in the Federal Republic of Germany (FRG), and the CO_2-cooled reactors in the United Kingdom such as Hunterson B and Hinkley Point B. The experience gained at each of these reactors has proved the high safety potential of GCR power plants.

Since the first criticality in 1974, the FSV has operated at power levels up to 70% and produced over 10^6 kWh electricity. During the startup phase, it has encountered problems of the type that would be expected in a first-of-a-kind system, such as steam generator leakage, purified helium leakage, and circulator bearing problems. All of these problems dealing with the secondary cooling and auxiliary circuit could be solved. Another problem, the periodic fluctuations in gas temperatures, has already been solved by installing region constraint devices (RCDs) that prevent gaps between regions and blocks. These gaps were assumed to be the reason for the fluctuations. Subsequent to the RCD installation, no fluctuations have been detected within the present power limit of 70%. However, the behavior of the plant at 100% has yet to be demonstrated.

The AVR has been successfully operated for more than 12 years. The helium-outlet temperature of 850°C was increased to 950°C in the spring of 1974. The inherent safety features of the AVR have been demonstrated several times by the shutdown of the primary cooling system during full-power operation. The reactor, left uncontrolled and without use of control rods, at once became subcritical. After 15 to 24 h, it became critical again and balanced at a power of <1% of the nominal power of 15 MW(electric). During this demonstration the fuel elements showed a maximum temperature increase of only 65°C.

The successful operation of the British advanced gas-cooled reactors (AGRs) has led to new construction plans for AGR plants in Scotland (Torness) and in England (Heysham). The following main design changes are planned for the new power plants:

1. increased number of channels
2. increased size of main boilers
3. increased size of the prestressed concrete reactor vessel (PCRV)
4. injection of nitrogen into discrete channels as an additional diversity of secondary shutdown system.

Another demonstrated advantage of GCRs is the low occupational radiation exposure. The favorable exposures are supported by the results from the Peach Bottom and FSV plants, as well as by operating experience of the

British Wylfa and Oldbury stations, which also have PCRVs. The data of average man-rem/MW(electric)·yr is a factor of 10 lower than the average data of all light water reactors (LWRs) in operation in the U.S.

For an increased understanding of nuclear power plants' response to postulated accident sequences at the Los Alamos National Laboratory, a whole plant simulation code (CHAP-2) is being developed. For the FSV the code includes 21 coded modules that model

1. the neutron kinetics and thermal response of the core
2. the thermal hydraulics of the reactor primary coolant system, secondary steam supply system, and balance of plant
3. the actions of the control system and plant protection system
4. the response of the reactor building and the relative hazard resulting from fuel particle failure.

First results of calculations were presented showing good agreement between code output and plant data.

The licensing experience of a prototype power plant, the Thorium Hochtemperatur Reaktor (THTR)-300 in the FRG, was described. Since this commercial plant is the first of its kind to be licensed and licensing procedures are governed by water-cooled reactors, time delays were unavoidable. Construction was started in 1972 and since that time the safety requirements have substantially increased.

Since the main components of THTR have already reached a state of design permitting only limited modifications, the plant protection system consequently had to be modified. The plant protection system consists of

1. decay heat removal system
2. reactor scram system
3. penetration closure
4. steam generator protection system.

The licensing procedure was demonstrated for the steam generators. Two steps have to be approved by the licensing authorities:

1. the licensing of the manufacture
2. the licensing of the system.

In some cases, the approval of the manufacture took such a long time that in the meantime new safety requirements were issued necessitating changes in the manufacture plan, which had to be approved again by the licensing authorities.

Two papers were written concerning the development of gas-cooled fast breeder reactors (GCFBRs).

The reference design of the European Association for the GCFBR has some installed safeguards that are different from other GCRs.

1. *Vented pin concept for the fuel elements*: The high helium pressure of ~90 bar necessitates vention of fission gases to a central fission product trapping system to eliminate differential gas pressure in the cladding. This design allows to operate even with minor cladding failures without circuit contamination.
2. *Design of the heat removal systems*: Three auxiliary loops (AHS) are always operating together with the main loops even under normal conditions. Three independent emergency cooling loops provided backup solution against a common mode failure of the main loop and AHS.
3. *Reactor shutdown system*: The shutdown rods of the second system are equipped with a melting device, which would release the absorber at coolant temperatures >800°C in event of failure of the normal trip actuation.

These specific GCFBR safeguards in combination with the PCRV and a necessary double shell containment essentially reduce the overall safety risk to a favorable level comparable with that of other reactor systems.

In the revised Gas-Cooled Fast Reactor (GCFR) Safety Program Plan of the U.S., a quantitative risk limit line has been adopted to establish requirements for the safety-related functions and systems. The risk limit line is derived from an interpretation of U.S. Nuclear Regulatory Commission established licensing requirements, including those for liquid-metal fast breeder reactors. Multiple barriers to the progression of accident sequences are defined in the form of six lines of protection (LOPs). LOPs-1 to -3 are dedicated to accident prevention and represent the normal operating systems, the dedicated safety systems, and the inherent design features, respectively. LOPs-4 to -6 are dedicated to the mitigation of core melt accident consequences and include in-vessel accident containment, secondary containment integrity, and radiological attenuation, respectively. Cumulative frequency limits and consequence limits are established for each LOP. Design features associated with each LOP were described.

Each of the LOPs defined in the plan provides an independent, sequential, and quantifiable risk barrier between the public and the potential radiological hazards associated with operation of the proposed GCFR demonstration plant. Reliability analyses and system design and performance studies completed to date indicate that GCFRs can be operated without imposing undue risk to the health and safety of the public.

SUMMARY OF SESSION II: SAFETY ANALYSIS AND RESEARCH

On the subjects of Safety Analysis and Research, the session dealt mainly with Safety Analysis, although with some emphasis on experimental validation of computing codes which are used.

The first two papers were concerned with probabilistic risk assessment. The first reviewed work at General Atomic Company, much of which had already been published. The paper indicated the scope of the work and also contains a useful and convenient compilation of references. Notable in the scope is the pioneering work done in five hazards analysis. Discussion brought out the close dependence of a careful assessment of common cause failure possibilities.

The second paper presented results of parallel work in Germany. The paper contains information on the results of this work, much of which is new. These results require quiet study outside the conference and their availability is welcome. Discussion noted differences between U.S. and German conclusions for which explanations were sought. More discussion along these lines could well be of interest but was not possible in the time at our disposal.

Another presentation from FRG considered a 350-MW(thermal) annular pebble bed aimed at making full use of inherent protection mechanisms and especially the natural removal of shutdown heat. Calculations were presented showing that conditions are acceptable even in the total absence of coolant flow; acceptable in the sense that although damage may well occur to the PCRV, it is considered that the consequences of this very severe fault can be fully contained.

Two papers showed the relative simplicity of calculating the pressure transient in the course of a depressurization fault, a situation that provides a marked contrast with the LWR situation. Both papers gave experimental results in very satisfactory agreement with the calculated predictions. One paper did indicate, however, that oversimplification of calculation methods can lead to more discrepant predictions.

The next paper presented some conclusions drawn from experience in licensing procedures for the THTR-300 and the HTR 1100 design, which was based on the use of multiblock-type fuel elements. After some difficulties due to the relative novelty of licensing procedures being applied to GCRs in Germany, criteria were established and analysis of the resulting design is considered to confirm the favorable safety characteristics that were anticipated. Points receiving particular considerations were depressurization and water ingress accidents. It was brought out in the discussion that it is now claimed that high quality manufacture and in-service inspection make it possible to exclude the possibility of failure of a complete tubesheet together with reactor circuit depressurization and that water ingress could be limited to that resulting from a single tube failure.

The next paper described the HTGR Safety Research Program pursued since 1974 in the U.S. by General Atomic. The program includes an investigation under unrestricted core heatup conditions of the following: fission product release and plate-out, PCRV integrity, containment atmosphere response, and criticality. The results provide fundamental support to the updates of HTGR probabilistic risk assessment study and are of essential importance for the U.S. HTGR licensing procedure. To reduce identified uncertainties, the future safety research priorities will include: PCRV top head degradation containment failure modes and long-term iodine deposition characteristics.

The aspect of water ingress accidents in HTRs and two aspects of air ingress were described. These aspects were the pressure buildup in the primary circuit in the case of water ingress accidents and the air ingress rate as well as the graphite corrosion in accidents involving air ingress.

Taking the THTR as an example, it was shown that the pressure buildup in the primary circuit is not critical even in the case when the localization and shutoff of the source is not successful.

In general it was shown that air ingress rates are very small in the case of a top leak of the PCRV and cause no danger to fuel and containment.

The often feared graphite fire can be excluded even if a high percentage of containment atmosphere is transported into the core region by the circulators of the afterheat removal.

One paper dealt with a detailed study of the steam entry in GCFR. Exclusive steam entry calculations have been performed for a detailed model of a 300-MW(electric) GCFR. The results show that for the middle of the equilibrium cycle and steam densities in the coolant channels in the range of interest the overall effect was computed to be negative; however, a thorough analysis demonstrated that the effect is very sensitive to the basic data, the geometric and nuclear data. Particularly interesting are the large negative contributions of the blanket zones to the total effect.

The transient analysis of an HTGR with gas turbine cycle was described in a paper from Switzerland. Two cases were presented: a generator loss of load with plant trip and a generator loss of load followed by plant standby operation. In both cases a compressor bypass must open to decrease the pressure ratio and to suppress the excess power of the turbine.

In addition to these cases, very severe pressure equalization transients initiated by unlikely events like the deblading of one or more turbo-machines must be taken into account.

The last paper contained the safety assessment of a multicavity PCRV with hot liner.

The PCRV makes it possible to increase the inherent safety of the plant by including in the vessel the whole of the thermal and nuclear equipment pertaining to the main circuit.

The PCRV of the HHT Project differs from other PCRVs by the important number of cavities, different cavity pressures, and by a liner in contact with hot gas. Due to the great thickness of the walls, it is important to design a drainage system. As possible linear material, a steel with low coefficient of thermal expansion is suggested.

The cavity closures are an important aspect of the vessel safety. Reinforced concrete shell with independent liner is proposed.

SUMMARY OF SESSION III: FUTURE PROSPECTS FOR GCR SAFETY

The GCRs achieve a high degree of plant safety relying on inherent features, such as single phase gas as coolant, PCRVs, ceramic core and reflector [exception: gas-cooled fast breeder (GCFB)] and coated particle fuel (exception: GCFB and AGR). These features are supplemented by engineered safeguards such as dedicated heat removal systems, diverse and redundant shutdown systems, and a secondary containment structure to prevent release of any radioactivity that may escape the primary coolant boundary. Engineered safety features provided for water ingress in the primary coolant include moisture detection systems, steam generator isolation and dump systems, and PCRV relief valves.

Additional features such as natural convection cooling loops may reduce even further the possibility of core heatup, which, however, is already extremely low for all present concepts.

The licensing experience of HTRs in the U.S. was described based on the Peach Bottom Atomic Power Station and FSV as nuclear plants, which have been or are still in operation, and on plants that were to be built, such as Philadelphia Electric 1160-MW(electric) plant, Delmarva 770-MW(electric) plant, and General Atomic 1160-MW(electric) standard plant.

The history of licensing of the HTGR in the U.S. has been one of changing requirements due to changing design concepts, changing plant size, and changing level of detail of the review by the regulatory authority. The influence of the Three Mile Island accident on the licensing of HTGR concepts was described as minimal. An increasing use of the probabilistic risk assessment is expected to result in heightened awareness of the safety margins inherent in the HTGR compared with other reactor types.

In the United Kingdom, the preliminary safety report and the preconstruction report have been submitted for two new AGR plants (Heysham II and Torness). Construction on site was scheduled to start in autumn 1980. Some design changes have been necessary for the new plants (compare Session I) since an interval of 12 years has passed since the Hinkley Point and Hunterson reactors were ordered.

The multipurpose very high temperature reactor (VHTR) with 50 MW(thermal) is being developed in Japan. The reactor will be designed for 1000°C gas outlet temperature. The reactor will be used for demonstration experiments of nuclear process heat application such as production of reducing gas for steel-making. The major design differences to other GCRs are:

1. intermediate helium/helium heat exchanger (IHX)

2. N_2 gas injection system

3. steel pressure vessel (for this experimental station).

The IHX enables a pressure difference between the primary and secondary helium circuit, which prevents fission product leakage into the secondary system.

The N_2 gas system prevents the core support graphite from being oxidized by the air ingressed after the primary pipe rupture accident.

In the USSR an HTGR (VGR-50) is being developed with thermal output of 136 MW and a helium pressure of 800°C. The produced energy will be used for chemical processes. Like the VHTR in Japan and the AVR in FRG, this experimental plant has no PCRV but a steel pressure vessel.

The analysis of the following accident conditions was reported:

1. 0.1% reactivity perturbation (i.e., as result of one graphite block dropping from the steam reflector on the core)

2. de-energization of four gas blowers

3. fuel circulation scram

4. water ingress

5. depressurization of the main duct.

Results proved that the VGR-50 is effectively self-controlled and has low sensitivity to the failure of the core cooling. The primary circuit depressurization accident was reported to be the most severe accident due to a possible subsequent core overheating. Therefore, it was recommended to investigate additional research and development work and theoretical calculations.

The last contribution concerned investigation of explosion in a nuclear coal gasification plant. The nuclear system must be protected against the impact of possible process gas explosions. The results of an extensive program have demonstrated that escaping process gases cannot detonate and that in the worst case their explosion process will be an enhanced deflagration.

APPENDIX

PAPERS PRESENTED AT THE MEETING

(Copies of papers may be requested from the authors)

G. C. BRAMBLETT, C. R. FISHER, and F. E. SWART, "Operational Experience at Fort St. Vrain," General Atomic Company, San Diego, California, United States of America

G. IVENS and E. ZIERMANN, "Safety Aspects of the Operation Experience with the AVR Experimental Power Station," Arbeitsgemeinschaft Versuchsreaktor AVR GmbH, Düsseldorf, Federal Republic of Germany

R. BREITENFELDER, W. WACHHOLZ, and U. WEICHT, "Accident Analysis and Accident Control for the THTR-300 Power Plant," Hochtemperatur-Reaktorbau GmbH, Mannheim, Federal Republic of Germany

H. W. FRICKER, "THTR Steam Generator Licensing Experience as Seen by the Manufacturer," Sulzer Brothers Ltd, Winterthur, Switzerland

R. M. YEOMANS, "Advanced Gas-Cooled Reactors (AGRs)," South of Scotland Electricity Board, Hunterston Power Station, West Kilbride, Ayshire, United Kingdom

J. CHERMANNE and P. BURGSMÜLLER, "Gas-Cooled Breeder Reactor Safety," European Association for the Gas-Cooled Breeder Reactor (GBRA), Belgonucléaire, Brussels, Belgium

A. TORRI and D. R. BUTTEMER, "Gas-Cooled Fast Reactor Safety–An Overview and Status of the U.S. Program," General Atomic Company, San Diego, California, United States of America

S. SU and B. A. ENGHOLM, "Personnel Radiation Exposure in HTGR Plants," General Atomic Company, San Diego, California, United States of America

K. R. STROH, "Developmental Assessment of the Fort St. Vrain Version of the Composite HTGR Analysis Program (CHAP-2)," Los Alamos Scientific Laboratory, Los Alamos, New Mexico, United States of America

K. N. FLEMING, W. J. HOUGHTON, G. W. HANNAMAN, and V. JOKSIMOVIC, "Probabilistic Risk Assessment of HTGRs," General Atomic Company, San Diego, California, United States of America

J. FASSBENDER and W. KRÖGER, "Results of a German Probabilistic Risk Assessment Study for the HTR-1160 Concept," Kernforschungsanlage Jülich GmbH, Jülich, Federal Republic of Germany

K. PETERSEN, R. BUSCHER, H. GERWIN, and W. SCHENK, "Efficiency of Inherent Protection Mechanisms for an Improved HTR Safety Concept," Kernforschungsanlage Jülich GmbH, Jülich, Federal Republic of Germany

G. FRITSCHING and G. WOLF, "Model Experiments on Depressurisation Accidents in Nuclear Process Heat Plants (HTGR)," INTERATOM, Internationale Atomreaktorbau GmbH, Bergisch Gladbach, Federal Republic of Germany

M. DANG, J.-F. DUPONT, and H. WEBER, "Rapid Depressurization of a Helium Vessel: A Comparison Between Experimental Data and One-Dimensional Analysis," Swiss Federal Institute for Reactor Research, Würenlingen, Switzerland

H. SOMMER and D. STÖLZL, "The HTR Safety Concept Demonstrated by Selected Examples," Hochtemperatur-Reaktorbau GmbH, Mannheim, Federal Republic of Germany

A. W. BARSELL, B. E. OLSEN, and F. A. SILADY, "HTGR Safety Research Program," General Atomic Company, San Diego, California, United States of America

J. WOLTERS, "Aspects of Water and Air Ingress Accidents in HTRs," Kernforschungsanlage Jülich GmbH, Jülich, Federal Republic of Germany

W. HEER, P. STILLER, H. U. WENGER, and P. WYDLER, "A Detailed Study of the Steam Entry Effect of a GCFR," Swiss Federal Institute for Reactor Research, Würenlingen, Switzerland

M. DANG, J.-F. DUPONT, P. JACQUEMOUD, and R. MYLONAS, "Pressure Transients Analysis of a High-Temperature Gas-Cooled Reactor with Direct Helium Turbine Cycle," Swiss Federal Institute for Reactor Research, Würenlingen, Switzerland

H. WEBER, "Investigation on the Depressurization Behaviour of a Typical HHT Thermal Insulation," Swiss Federal Institute for Reactor Research, Würenlingen, Switzerland

R. LAFITTE and J. D. MARCHAND, "Safety Assessment of a Multicavity Prestressed Concrete Reactor Vessel with Hot Liner," Bonnard and Gardel Ing. Conseils, Sa., Lausanne, Switzerland

C. R. FISHER and D. D. ORVIS, "Licensing of HTGRs in the United States," General Atomic Company, San Diego, California, United States of America

J. M. YELLOWLEES and E. C. COBB, "Safety Design Features for Current U.K. Advanced Gas-Cooled Reactors," Nuclear Power Company Ltd., Risley Warrington, Cheshire, United Kingdom

T. YASUNO, S. MITAKE, M. EZAKI, and K. SUZUKI, "The Reactor Safety Study of Experimental Multi-Purpose VHTR Design," Japan Atomic Energy Research Institute, Tokai Research Establishment, Tokai-Mura, Naka-Gun, Ibaraki-Ken, Japan

V. JOKSIMOVIC and C. R. FISHER, "HTGR Safety Philosophy," General Atomic Company, San Diego, California, United States of America

V. N. GREBENNIK, E. I. GRISHANIN, N. E. KUKHARKIN, P. V. MIKHAILOV, V. V. PINCHUK, N. N. PONOMAREV-STEPNOY, G. I. FEDIN, V. N. SHILOV, and I. V. YANUSHEVICH, "Analysis of Some Accident Conditions in Confirmation of the HTGR Safety," Institute of Atomic Energy P. L., I. V. Kurchatova, Moscow, USSR

M. S. BELYAKOV, V. D. KOLGANOV, S. S. KOCHETKOV, V. P. SMETANNIKOV, and V. K. ULASEVICH, "Use of Natural Circulation Mechanism for Core Cooling of High Temperature Helium-Cooled Reactors as a Means of Safety Enhancement," Institute of Atomic Energy P. L., I. V. Kurchatova, Moscow, USSR

K. SCHULTE, "Investigations of Gas Explosions in a Nuclear Coal Gasification Plant," INTERATOM, Internationale Atomreaktorbau GmbH, Bergisch Gladbach, Federal Republic of Germany