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Summary

This report presents the detailed study of safety features and defines safety requirements of high temperature reactors with peeble bed core (chosen in WP4) for the demonstrator program of a co-generation system, especially: safety realted systems of HTGR plant, pebble bed high temperature reactor fuel, plant response to postulated initiating events, fission product releases, fuel performance and fission product behavior under oxidizing conditions, regulations regarding radiological consequences of postulated accidents and dource terms and radiation doses at exclusion area boundary.

Approval

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ACRONYMS

AF	attenuation factors
AGR	Advanced Gas Reactor
D/Q	deposition
DBA	design basis accidents
EAB	exclusion area boundary
EPA	Environmental Protection Agency
HPB	helium pressure boundary
HTGR	high temperature gas-cooled reactor
LBE	licensing basis events
LPZ	low population zone
MACCS2	MELCOR Accident Consequence Code System
MHTGR	modular high temperature gas-cooled reactor
MST	Mechanistic Source Terms
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
PAG	Protective Action Guides
R&D	research and development
PBMR	Pebble Bed Modular Reactor
RB	reactor building
RCCS	reactivity cavity cooling system
ROT	reactor outlet temperature
SSC	safety-related systems, structures, and components
TRISO	tristructural-isotropic
χ/Q	atmospheric dispersion

1 Experience with pebble bed reactors

1.1 Early steel vessel HTGR prototype plants

1.1.1 Early HTGR plant designs

The early HTGR plant designs (Dragon, AVR and Peach Bottom) utilized steel primary system vessels. The general performance of these plants was very good, and they provided an excellent basis for further development of the HTGR.

The Dragon Reactor Experiment in the UK first operated at power in July 1965, and reached its full capability of 20 MW(t) by April 1966. Initiated in April 1959 as an international project of the European Organisation for Economic Co-operation and Development, the primary objective of this plant was to demonstrate the feasibility of the HTGR and to launch the development of a technology which had already begun at a low level in various national laboratories. The reactor incorporated graphite fuel elements with high enriched uranium-carbide coated fuel particles. The core exit and inlet helium temperatures were 750°C and 350°C, respectively. The plant operated for long periods at full power and demonstrated successful performance of many components. Although it did not produce electric power, it served as a productive research tool for the development of helium gas cooled reactors and advanced fuel particle coatings.

In Germany, the 15 MW(e), Arbeitsgemeinshaft Versuchsreaktor (AVR), a pebble bed type HTGR began operation in 1967. The AVR had a steel containment vessel and used particle fueled, graphite spheres 6 cm in diameter that traveled downward through the core.

Although the AVR initially included a core outlet temperature of 850°C, this was subsequently raised to 950°C without decrease in plant performance. The AVR was the main fuel development tool for the pebble bed concept, and it and supplementary laboratory fuel testing became the major support of Germany's position that an LWR-type containment barrier was not needed for future HTGRs. The operation of the AVR continued until December 1988. In completing 21 years of service, the plant had accumulated more than 122,000 hours of operation with a 66.4% overall availability and had generated 1.67 billion kW•h of electricity.

Although several GCRs were built in the USA during the 1950s and 1960s, mainly for propulsion purposes, the helium cooled, 40 MW(e) Peach Bottom HTGR provided the major background for the continued commercialization of HTGR technology in the USA.

Operated by the Philadelphia Electric Company from 1967 to 1974, the plant achieved a gross capacity factor of 74% and an overall availability of 88% (exclusive of planned shutdown for R&D programmes). This HTGR introduced coated particle fuel and initiated the development of licensing criteria for this type of GCR. Two reactor cores were utilized in the operation of Peach Bottom 1. The fuel particles of core 1 were coated with a single layer of anisotropic carbon to prevent hydrolysis of the carbide fuel kernels. Fast neutron induced dimensional changes resulted in fracture and distortion of the coatings, which eventually resulted in 90 cracked fuel element sleeves out of 804. Plant operation, however, was not impaired, and primary circuit activity reached a maximum of 270 Ci, well below the design activity level of 4225 Ci. Core 1 accumulated 452 equivalent full power days before it was replaced with core 2. core 2, which provided buffer isotropic pyrolytic carbon (BISO) coatings on the fuel particles, operated with no fuel failures for its full design life of 897 equivalent full power days. The primary circuit activity averaged only .5 Ci during this entire period. Throughout the operation of Peach Bottom 1, excellent agreement was found between the predicted and actual plant design characteristics, verifying the methods used and providing a reference data base for application to larger HTGR plants.

*1.1.2 **Prestressed concrete reactor vessel demonstration power plants***

The follow-on HTGR plants, Fort St. Vrain (FSV) and the thorium high temperature reactor (THTR-300), both featured primary systems enclosed in prestressed concrete reactor vessels (PCRVs). These plants came on line at a time when the commercial industry was in decline, limiting the resources available to resolve issues associated with the commissioning and operation of new power plant designs.

The FSV HTGR was operated by Public Service Company of Colorado as part of the United States Atomic Energy Commission's Advanced Reactor Demonstration Program. This plant was designed with several advanced features including: a) a prestressed concrete reactor vessel (PCRV) containing the entire primary coolant system, b) a core of hexagonal, graphite block fuel elements and reflectors with fuel in the form of ceramic coated (TRISO) particles, c) once-through steam generator modules producing 538°C superheated main and reheat steam, and d) steam-turbine-driven axial helium circulators. Although ~5.5 billion kW•h of electricity was generated at FSV, the plant operated at low availability primarily because of excessive downtime due to problems with the water-lubricated bearings of the helium circulators. In spite of this low availability, the plant was a valuable technology test-bed, successfully demonstrating the performance of several major systems, including the reactor core with TRISO coated fuel particles in hexagonal graphite blocks, reactor internals, steam generators, fuel handling and helium purification.

The THTR-300 power plant was sponsored by Germany (FRG) and Nordrhein Westfalen (NRW). Construction of this 300 MW(e) plant began in 1971, but primarily due to increasing licensing requirements, the plant was not completed until 1984. This pebble bed reactor plant was connected to the electrical grid of the utility Hochtemperatur-Kernkraftwerk GmbH (HKG) in November 1985. In August 1989, the decision was made for the permanent shutdown of the THTR-300. This action was not due to technical difficulties associated with the plant, but was the result of an application by HKG for early decommissioning based on a projected short fall in funding and contractual changes in the allocation of decommissioning costs between the FRG, NRW and HKG that would take effect upon the termination of the demonstration phase in 1991. Operation of the THTR-300 was successful in validating the safety characteristics and control response of the pebble bed reactor, primary system thermodynamics and the good fission product retention of the fuel elements.

*1.1.3 **The AVR pebble bed reactor***[1]

The AVR pebble bed reactor (46 MWth) was operated 1967-88 at coolant outlet temperatures up to 990°C. A principal difference of pebble bed HTRs as AVR to conventional reactors is the continuous movement of fuel element pebbles through the core which complicates thermohydraulic, nuclear and safety estimations . Also because of lack of other experience AVR operation is still a relevant basis for future pebble bed HTRs and thus requires careful examination.

In the AVR the steam generator of pressure equal 73 bar was inside the reactor vessel.

As hot gas temperatures of up to almost 1000°C were achieved in AVR, which allow for process heat applications, the pebble bed technology finds major interest worldwide .

*1.1.4 **AVR safety issues***

AVR safety issues were presented in a paper of Moormann[2], which deals mainly with some insufficiently published unresolved safety problems of AVR operation and of pebble bed HTRs and skips the widely known advantageous features of pebble bed HTRs.

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[1] A safety re-evaluation of the AVR pebble bed reactor operation and its consequences for future HTR concepts Rainer Moormann Berichte des Forschungszentrums Jülich ; Jül-4275, 2008

[2] Rainer Moormann: ibid

For average AVR hot gas temperatures of 950°C maximum surface temperatures of fuel elements were originally calculated to 1070°C.

Depending on the fission power of the respective fuel element, maximum coated particle temperatures are up to 120 K higher . Modern fuel elements were designed for a maximum coated particle temperature in long term normal operation of < 1250°C . However sufficient irradiation tests were until recently available only for lower temperatures or low burn-up

In Moormann's paper, the term core temperature means fuel element surface temperature.

The AVR primary circuit is heavily contaminated with metallic fission products (Sr-90, Cs-137) which create problems in current dismantling. The amount of this contamination is not exactly known, but the evaluation of fission product deposition experiments indicates that the end of life contamination reached several percent of a single core inventory, which is some orders of magnitude more than precalculated and far more than in large LWRs . A major fraction of this contamination is bound on graphitic dust and thus partly mobile in depressurization accidents, which has to be considered in safety analyses of future reactors . A re-evaluation of the AVR contamination is performed here in order to quantify consequences for future HTRs (400 MWth). It leads to the conclusion that the AVR contamination was mainly caused by inadmissible high core temperatures, increasing fission product release rates, and not – as presumed in the past - by inadequate fuel quality only.

The high AVR core temperatures had not been detected until one year before final AVR shut-down, because a pebble bed core cannot yet be equipped with instruments. The maximum core temperatures are still unknown but were more than 200 K higher than calculated. Further, azimuthal temperature differences at the active core margin of up to 200 K were observed, probably due to a power asymmetry. Unpredictable hot gas currents with temperatures > 1100°C, which may have harmed the steam generator, were measured in the top reflector range.

After detection of the inadmissible core temperatures, the AVR hot gas temperatures were strongly reduced for safety reasons. Thus, a safe and reliable AVR operation at high coolant temperatures, which is taken as a foundation of the pebble bed VHTR development in Generation IV, was not conform with reality. Despite the remarkable effort spent on this problem, the high core temperatures, the power asymmetry and the hot gas currents are not yet understood. It remains uncertain whether convincing explanations can be found on the basis of poor AVR data and whether pebble bed specific effects are the reasons. Respective examinations are however ongoing.

Reliable predictions of pebble bed temperatures are at present not yet possible. The AVR contamination problems are related to the fact that even intact HTR fuel elements do not act as an almost complete barrier for metals, as they do for noble gases. Metals diffuse in fuel kernel, coatings and graphite and their break through takes place in long term normal operation, if fission product specific temperature limits are exceeded. This is an unresolved weak point of HTRs and is in contrast to other reactors. Intact LWR fuel elements represent a complete barrier despite of fuel centre temperatures reaching up to 2500°C, because claddings remain at temperatures < 600°C which excludes release by diffusion . Another disadvantage of HTRs, responsible for the pronounced contamination, lies in the fact that activity released from fuel elements is distributed in HTRs all over the coolant circuit surfaces and on graphitic dust and accumulates there . Deposition rates of chemical reactive fission products in the HTR coolant circuit are large. Thus, the removal of activity released from core by a coolant purging facility like in LWRs cannot be performed in gas cooled reactors.

1.1.5 ***Consequences of AVR experience on future reactors.***

The influence of fuel quality on the AVR contamination was examined by Moormann[3]. In contrast to Sr and Ag the retention of Cs in intact coated particles of modern TRISO fuel, as present in AVR during its final years of operation, is even worse compared to former HTI-BISO fuel . On the other hand the fraction of defective fuel particles and of uranium outside particle kernels is smaller in modern fuel.

Further, the retention of Sr in oxides kernels used in modern fuel is better than in former carbide kernels. For an AVR operation with only modern TRISO fuel the contamination is estimated to be by a factor of 10 to 30 lower for Sr-90 . Smaller reductions are expected for Cs and Ag. These results are not in conflict with recent high temperature irradiations of modern fuel, which discovered significant higher activity releases than expected.

As long as pebble bed immanent reasons for high core temperatures cannot be excluded they have to be conservatively considered in operation and design basis accidents of future pebble beds HTRs. For that case we have to note that AVR was operated for only less than 4 y at hot gas temperatures > 900°C, and thus primary circuit contaminations in future reactors (400 MWth, 900°C hot gas temperature, modern fuel, 32 full power years) are expected to approach at least the same order as in AVR end of life. This creates problems: Former safety analyses for advanced small HTRs revealed that activities accumulated in the primary circuit are a major source term contributor in design basis accidents . Further maintenance and dismantling is significantly hindered. Another consequence of inadmissible high core temperatures to be considered in future reactors is the transgression of temperature limits, which prevent from formation of explosive gas mixtures in water ingress accidents of steam cycle and certain process heat generating designs . Ingress of liquid water into the pebble bed, as it accidently happened in AVR, has to be excluded in future reactors by design measures in order to avoid a potential positive void coefficient of reactivity with reactivity transients.

Criteria for the maximum tolerable accumulated activity in the coolant circuit are developed on basis of German regulations for protection of the public, i.e. maximum tolerable releases in design basis accidents, and of requirements from maintenance and disposal. Application of these criteria on advanced pebble bed reactors leads to the conclusion that a pebble bed HTR needs a gas tight containment even if inadmissible high temperatures as observed in AVR are not considered. However, a gas tight containment does not diminish the consequences of the primary circuit contamination on maintenance and dismantling.

Inadmissible high temperatures substantially aggravate these problems. A safe operation at hot gas temperatures near to those suitable for process heat applications can currently not be guaranteed by pebble bed reactors, even if a gas tight containment is present.

Thus complementary measures to a containment in order to achieve safe operation of pebble bed reactors despite of activity accumulation in the primary circuit and of temperature uncertainties are discussed. A reduction of demands on future reactors (hot gas temperatures, fuel burn-up) is one option; another one is an elaborate R&D program for solution of the following problems related to operation and design basis accidents:

- development of a new fuel element sufficiently retaining metallic fission products in long term operation. For hot gas temperatures as in process heat applications the retention of non metallic fission products has to be improved, too
- development of a reliable quality control for fuel elements
- experiments on iodine release from fuel elements in core heat-up accidents

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[3] Rainer Moormann: ibid

- full understanding and reliable modeling of core temperature behaviour and of pebble bed mechanics including pebble rupture
- fast and reliable local measurement (direct or indirect) of safety relevant parameters in the pebble bed core (e .g . temperatures)
- full understanding of fission product transport in the coolant circuit, including development of measures to avoid the current uncontrollable activity accumulation in the circuit
- development of a fast detection system for metallic fission product release from core
- material development for process heat components

*1.1.6 **HTR specific dismantling and disposal items***

AVR undergoes dismantling, which became complicated due to a pronounced contamination of the primary circuit with Sr-90 and Cs-137. Particularly highly radiotoxic Sr-90 creates safety concern because this contamination is partly present in mobile, dust borne form. Accordingly, the reactor vessel will be grouted with light concrete which immobilizes dust and stabilizes the vessel, and will be stored for some decades outside of the AVR site, until a procedure for final dismantling is developed.

A voluminously instrumented experimental pebble bed reactor would be required for solution of these problems . Before initiation of this comprehensive R&D a feasibility study including an estimate of the required effort is advisable in order to quantify the economical risk of this development. Comparative probabilistic safety assessments on pebble bed HTRs, HTRs with block type fuel and Generation III LWRs are proposed in order to generate a reliable figure of current pebble bed reactor safety[4]:

*1.1.7 **Re-evaluation of fission product release from AVR core into the coolant circuit***

Measurements of Cs, Sr and Ag release from AVR core by a deposition tube in the hot gas region (VAMPYR-I) revealed, that the fission product release into the primary circuit strongly accelerated 1974 – 1976, correlating with the hot gas temperature increase from 850 to 950°C in February, 1974 Considering AVR volume flow and underestimation of release rates by VAMPYR-I the core release rates R of Cs-137 were calculated from coolant activities A to be

$$R\ [Bq/s] = 500 \cdot A\ [Bq/m3]$$

The core release rate of Sr-90 for a core composition as in the time period 1974-78 at a hot gas temperature of 950°C was calculated to 20 GBq/y by standard diffusion models/data, i .e . several orders of magnitude smaller than observed . Unfortunately, the enhanced release of metallic fission products was detected with several months of delay only, when already a major contamination had happened: This was, because a fast release measurement of metallic fission products did not exist . In addition -in contrast to expectations- no conspicuous release of easily detectable noble gases by fuel failure accompanied the release of metallic fission products . Further, calculations of the total primary contamination on basis of VAMPYR-I data were erroneous until 1988 and underestimated the Cs-137 contamination by a factor of 5. Accordingly, during reactor operation the contamination was not taken as serious as it was and no consequences were drawn.

An important consequence of the large primary circuit contamination, which is mainly found in loose and adhesively bonded dust, is the amplification of AVR dismantling costs. Another consequence was discovered 1994 : In course of a slow accidental steam/water ingress of 2.75·104 kg in 1978, about 1500 GBq of the accumulated Sr-90 was washed out, together with 105 GBq of H-3 . Some of the water contaminated with Sr-90 and H-3 leaked by human error into the reactor grounding, from where

[4] Rainer Moormann: ibid

it reached the surrounding soil. The soil contamination on the AVR site ranges from 1 to 1200 Bq Sr-90/kg. Because the steam generator leak was small, this accident was not a design basis accident: Core temperatures were already low when large water amounts were present and thus the extent of the graphite/steam reaction remained limited.

1.1.8 Influence of unintentional temperatures on fission product release

Assuming that maximum temperatures measured in 1986-87 are not higher than in 1974-76 it becomes clear, that the enhanced fission product release is correlated to overheating : Hot gas temperatures of 850°C led to release rates by 2 - 4 orders of magnitude smaller. This conclusion is supported by measurements of fission product release in the US-Peach Bottom HTR with well known, continuously measured core temperatures (operated 1967 – 74 with block type fuel, representing a similar fuel development stage in its core 2 as AVR 1970-78):

Particularly the Sr release was by several orders of magnitude smaller than in AVR, but also Cs-release in Peach Bottom was substantially lower than in AVR.

The fraction of defective coated particles in Peach Bottom core 2 was even at 3 .4 % due to a high density of particles in the matrix . Average coated particle defect fractions in AVR varied with time but were smaller. In the period 1974 - 76 a limited increase of iodine and of noble gas release rates was observed in AVR . This enhanced release cannot be attributed to intact particles. The observed increase is however partly due to accelerated diffusion out of defective particles by higher temperature. An increase of the average coated particle failure fraction by a factor of 2 to 4 is estimated from that release data, which obviously cannot explain the release enhancement of metallic fission products.

This underlines that a strong correlation between coated particle defect fraction and release of metallic fission products does not necessarily exist.[5]

For completion it has to be noted, that the average AVR hot gas temperature was in 1976 due to a calculation error for several months even at about 990°C, which accelerated release of metallic fission products.

1.1.9 Release mechanisms of metallic fission products

We have to distinguish 3 sources of fission products : Intact coated particles, particles with defective coatings and the uranium contamination of the graphite resulting from manufacturing . Progress of fuel element technology (e .g . SiC coating) diminished the fraction of uranium in defective particles and uranium contamination during AVR operation from 10^{-2} to 10^{-4} . Non metals are virtually completely retained by intact particle coatings, i.e. the fuel element acts as an efficient barrier and a partial release occurs from defective particles and uranium contamination only. Metallic fission products however diffuse even through intact coatings and the latter become penetrable at high temperatures. For that, the fuel element is a sufficient barrier for metallic fission products only up to a certain temperature limit, which depends on mobility of the metal and on the irradiation time . For temperatures below these limits, i .e . negligible release rates from intact coated particles, the release of Cs and Ag is caused by the small level of uranium contamination and defective coated particles.

Diffusion through barriers is characterized by 2 parameters : The breakthrough time tB reflects the initial diffusion phase before steady state diffusion is reached and characterizes the maximum retention time of a barrier:

$$tB = l^2/(6D)$$

[5] Rainer Moormann: ibid

with l = coating thickness, D = diffusion coefficient of fission product in coating

Steady state diffusion rates in coatings are proportional to D/l . Because of the temperature dependence of diffusion in solids the breakthrough time decreases, and the stationary diffusion rate increases with increasing temperature . Besides temperatures there are other parameters accelerating diffusion rates in HTR fuel, as burn-up and neutron fluence . However, these parameters are not fully understood.

The following table 1.1 contains the breakthrough time at 1250°C (maximum coated particle design temperature) for different barrier components of HTR fuel pebbles . Buffer layer and LTI-PyC are not considered, because their retention capability is small compared to other barriers (except for the unique LTI-BISO coating in fuel type GLE-1) . The data scatter of about one order of magnitude reflects uncertainties of diffusion coefficients. The breakthrough time can be only approximately used as an indicator for the start of a significant fission product release: Because of the continuous increase of long lived fission product concentrations in normal operation tB represents a lower limit for break through, if within of the break through time tB the fission product concentration in the kernel sufficiently increased. On the other hand, for t = tB already some release has taken place, which depends on the ratio of coating thickness to kernel diameter. This release is for coated particles already at several percent.

Table 1.1: Breakthrough time tB [d] for metallic fission products at 1250°C in components of HTR fuel pebbles[6]

Nuclide	**Breakthrough time t B [d] in**		
X	HTI-PyC (85 µm, BISO)	SiC (35 µm, TRISO)	A3 matrix graphite. (5 mm)
Sr-90	1 - 10	10 - 100	< 200
Ag-110m	1 – 10	10 - 100	< 0.5
Cs-137	1000 - 10000	50 – 500	1

Additional diffusion calculations indicate that in oxide kernels the Sr-retention is dominated by interaction with the kernel, which has to be considered in addition to the data of table 1.1 . In contrast the graphite is the dominant Sr retention barrier for carbide fuel kernels.

Chemisorption of metallic fission products in matrix graphite, which is modelled by a partition coefficient at the graphite/gas boundary, also increases the retention for Sr .

Modern TRISO fuel strongly reduces the uranium contamination of graphite and thus the release rates of iodine and noble gases in normal operation .

A recent irradiation of 5 modern German fuel elements at high temperatures discovered a worse behaviour than expected . GLE-4 LEU fuel elements representing the highest quality achieved in the German HTR fuel program were used . GLE-4 fuel elements (1 g U-235, 6 g total heavy metal) were also present in AVR during its final operation. The experiment was undertaken in order to demonstrate the applicability of present HTR fuel for Very High Temperature Reactor (VHTR) applications. Experimental conditions were as follows: Maximum pebble central temperature 1250°C, average

[6] Rainer Moormann: ibid

temperatures in the particle containing zone about 1150°C, pebble surface temperatures 1000 – 1050°C, irradiation time about 1 y, burn-up 11 .1 % ***fima***[7]. It has to be noted that the burn-up target value of 15 .4 % fima was not reached . Even though the number of defective particles increased by a factor of about 10 more during irradiation than expected and also a "significant" release of Cs and Ag out of the fuel elements was observed . The latter remains to be quantified . During this experiment there was by human error a temperature excursion of about 250 – 300 K for a short period, which however was not expected to be responsible for the large activity release.

Short term temperature excursions in earlier irradiations (FRJ2-K15/3) resulted in an increased noble gas release during the enhanced temperature period only. An ongoing similar experiment with GLE-4 fuel elements at about 150 K lower temperatures does not show up to now higher noble gas releases than expected . These results underline that even modern HTR fuel is not yet suitable for high temperature applications, and that the maximum design temperature of fuel particles of 1250°C for current TRISO fuel is too optimistic and should be reduced by about 100 – 150 K.

*1.1.10 **Interpretation of AVR release rates after hot gas temperature increase to 950°C***

The curves of fission product releases versus time show that the temperature increase starting February, 1974 leads with a delay of about 0.5 to 1.5 y to release rates of Sr and Cs by about 3 orders of magnitude larger than before. The Ag-110m release became even almost complete soon after temperature increase and thus its increase is smaller. The delayed start of the release is mainly due to diffusion effects, but a slowly increasing particle failure fraction has contributed for Cs, too .

Accident tests of irradiated fuel elements revealed a reasonable Sr and Cs (but not Ag) retention for TRISO oxide fuel in the short term (50 – 1000 h) for low and medium burn-up at temperatures up to 1600°C : Cs-release for medium burn-up fuel starts after about 40 – 200 h and amounts to about 1 % after 1000 h heating [8].

*1.1.11 **Maximum permissible environmental release in design basis accidents***

What are the implications of temperature uncertainties for future pebble bed HTRs?

Enhanced fission product accumulation in the primary circuit during normal operation is a major safety concern, because these activities were found by safety analyses to be main source term contribution in design basis accidents, and may even contribute significantly to the risk.

Future HTRs should be designed with a gas tight containment, as the AVR had.

Table 1.2 contains release limits of individual key nuclides into the environment in case of design basis accidents for German licensing conditions . The values given are calculated on basis of the maximum tolerable doses (50 mSv effective and red bone mark dose, 150 mSv thyroid dose). These values are lower than the values prescribed in the USA (250 mSv whole body dose, and 3 Sv thyroid dose), and much higher than the dose limits in force in Poland (10 mSv TEDE[9])

For a given source term there are only two site specific parameters, which influence the doses at the fence: These are the minimum distance to the fence and the emission height . In case of a common release of several nuclides, the individual exclusion release limits drop.

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[7] Fima - fissions per initial metal atom

[8] Rainer Moormann: ibid

[9] TEDE – total effective dose equivalent

Further on, ALARA holds here and therefore the release limits must not be reached, if that is reasonably achievable. This table contains equilibrium core inventories of the HTR-Module200 as a typical example for a modern design with advanced fuel, too.

Tab. 1.2 Calculated release limits of single nuclides into the environment for design basis accidents (derived from German regulations, distance to site fence: 100 m; release duration: 8 h)[10]

Nuclide	Half life	Total core inventory HTR-Module (200 MWth) [GBq]	Release limit [GBq] for emission height	
			20 m	50 m
Sr-90	28.8 y	$1.37 \cdot 10^7$	0.4	0.6
Ag-110m	250 d	$1.89 \cdot 10^5$	270	410
I-131	8 d	$2.07 \cdot 10^8$	5.5	10
Xe-133	5 .3 d	$4.44 \cdot 10^8$	$5.7\ 10^7$	$1.1\ 10^8$
Cs-137	30.1 y	$1.67 \cdot 10^7$	30	50

From data in table 1.2 the maximum tolerable mobile fraction of fission products accumulated in the primary circuit can be estimated from the point of view of protection of the public. In case that unfiltered releases into the environment in design basis accidents (i.e. no gas tight containment), a conservatively estimated mobile fraction of accumulated activities must remain sufficiently below the evaluated exclusion release limits of table 1.2. In presence of a gas tight containment with filtered release an accumulated mobile activity by at least 2 orders of magnitude larger than exclusion limits of table 1.2 becomes tolerable.

In presence of a gas tight containment other limiting factors may become effective, as there are the above discussed requirements of maintenance and of dismantling:

Restrictive limits are expected also for a complete exchange of the side reflector.

Metallic fission products occur mainly dust borne in the coolant circuit of pebble bed reactors. The dust is deposited by gravitation in dead water regions or by adhesive forces on surfaces. The mobilization of deposited activity in pebble bed reactors in course of accidents is not well examined, but some effort is at present spent into that item . There are however data on activity mobilization of specimen from Peach Bottom HTGR . These specimen were similar to AVR and THTR covered with a carbonaceous layer and contained more than 80 % of Cs and Sr released into the coolant circuit of the Peach Bottom HTGR. Blow down tests revealed that at shear forces by a factor of 5 larger than in normal operation between 2 and 25 % of the deposited activity is released within 2 minutes. Later tests on specimen without a carbonaceous layer did not show a mobilization of deposited metallic fission products, but a small release of iodine . For conservative estimations as required in design

[10] Rainer Moormann: ibid

basis accidents a mobilization degree of 25 % is assumed for depressurization accidents of pebble bed reactors . Japanese estimates result in a release of 15 % of Cs and 40 % of the iodine deposited in the coolant circuit. Formation of carbon dust cannot yet be avoided in current pebble bed reactors.

1.1.12 ***Consequences of AVR experience for water ingress accidents***

Another major safety relevant implication of inadmissible high core temperatures lies in the fact that the formation rate of burnable gases in design basis water ingress accidents increases exponentially with temperature : A mixture of CO and H2 is formed by interaction between steam and graphite . This problem occurs in steam cycle and in process heat generating concepts without an intermediate heat exchanger which is the expected arrangement for the pebble bed reactor to be installaed in Poland for industrial use. A primary circuit depressurization is hardly avoidable in this accident sequence . In order to prevent explosive gas mixtures after depressurization, maximum graphite surface temperatures must not exceed 1100 - 1200°C, depending on the core temperature distribution . This corresponds to maximum tolerable hot gas temperatures of < 750°C, if maximum core temperatures of 250 K higher than calculated with standard methods and as occurring in AVR are assumed . The presence of burnable graphite dust in pebble bed reactors may worsen the situation . An enlarged fission product release into the primary circuit followed by water ingress, remobilization of the accumulated activity, depressurization and destruction of last barriers by a gas explosion cannot be accepted as a potential scenario for a design basis accident . Moreover, such accidents are expected to proceed fast, i.e. within of 10 – 20 min, . Accordingly, emergency measures become difficult to perform.

The prevention of these accidents may require major design changes as reduction of temperatures or an explosion proven or inertized containment.

1.1.13 ***Conclusion***

The following major problems of pebble bed HTRs were identified during a re-evaluation of the safety behaviour of the AVR operation:

- Inadmissible high core temperatures, which heavily accelerated the activity release from fuel elements into the coolant circuit, and whose reasons are not yet understood

- An insufficient retention capability of present TRISO fuel elements for metallic fission products particularly in high temperature, long term normal operation as required for process heat applications.

- Safety, maintenance and disintegration problems due to the uncontrolled accumulation of metallic fission products all over the primary circuit. Fission products are present in the circuit to a large part in a mobile form.[11]

1.2 Modular pebble bed reactor HTR-10

The structure of modular pebble bed reactor HTR-10 is described in the publication of 2002[12].

1.2.1 ***Introduction***

The 10 MW High Temperature Gas-cooled Test Reactor (HTR-10) is a modular pebble bed type reactor.

The Modular High Temperature Gas-cooled Reactor concept was proposed originally as the 'HTR-Module' by Siemens German in 1979. The most important design features are as follows:

[11] Rainer Moormann: ibid

[12] Zongxin Wu *, Dengcai Lin, Daxin Zhong The design features of the HTR-10 Nuclear Engineering and Design 218 (2002) 25–32

- The use of spherical elements, which are capable of retaining all radiologically relevant fission products up to fuel element temperatures of approx. 1600 °C.
- The reactor core is designed and laid out such that a maximum fuel element temperature of 1600 °C is not exceeded during any accident.
- Active core cooling is not necessary for decay heat removal during accidents. It is quite sufficient to discharge the decay heat by means of passive heat transport mechanisms (such as heat conduction, radiation, nature convention) to a simple cavity cooler outside the reactor pressure vessel.
- Reactor shutdown is carried out solely by absorber elements, which, on demand, can drop freely into borehole of the reflector.
- Graphite is used in core areas with high temperatures (fuel elements, core internals). Temperature-incurred failure of these materials is impossible at the maximum occurring temperature of 1600 °C.
- The noble gas helium, which is neutral from a chemical and neutron physical viewpoint, is used as coolant.
- Due to the high activity retention of the fuel elements, a pressure-tight reactor building is not necessary. The reactor building is accessible for repair work at any time after an accident as a result of the low activity release.
- Reactor core and steam generator are installed in separate steel pressure vessels in such a way that there is no danger of components overheating in the case of failure of primary circuit cooling. This chosen installation also increases the accessibility of the components for maintenance and repair.

The Modular HTGR is well known for its excellent safety in terms of the following features:

- Capacity of retaining all fission products in the coated particles up to the temperature of 1600 °C.
- Large fuel temperature margin and negative temperature reactivity coefficient sufficient to accommodate reactivity insertion.
- Large heat capacity of the core to mitigate temperature transition.
- Small amount of excess reactivity in the core.
- Passive decay heat removal.

The core of the HTR-10 is formed as a pebble bed with 27 000 spherical fuel elements. The pebble bed reactor is considered for its merit of continuous discharge of spherical fuel elements, which can increase the reactor operation availability.

Spherical fuel elements with ‘TRISO’ coated particles are used in the HTR-10. The UO fuel kernel of particles is coated by ceramic layers, namely the inner buffer layer with less dense pyro-carbon, the dense pyro-carbon, the SiC coating and the outer layer of dense pyro-carbon. A spherical fuel element of 60 mm diameter is composed of an inner zone of 50 mm diameter and 5 mm thick shell of fuel free zone and about 8000 coated fuel particles are dispersed in the graphite matrix of the inner zone.

In any severe accident conditions the decay heat of the HTR-10 could be transferred and dissipated into the atmosphere via the passive decay heat removal system. The maximum temperature of the fuel should not exceed the temperature limit of 1600 °C, below which the SiC layer could retain the fission products within the particles, in the accident transients. Therefore the core damage accidents could be excluded in the Modular HTGR.

In terms of the requirement of HTR-10, the fraction of uncoated free uranium content of the fuel elements should be less than 8×10^{-4} during manufacture and irradiation. Then the gaseous fission products of the uncoated free uranium could be released into the primary coolant system and the helium purification system would remove most of the fission products from the primary coolant and maintain the concentration of fission products in the primary coolant at a low level.

Once the accidents of pipe rupture have happened, the loss of coolant and depressurization will occur and the fission products in the primary coolant will be released into the atmosphere. The accident analyses indicate that the impact of the radioactivity release on the environment would be much lower than the regulatory limits set for the HTR-10.

*1.2.2 **Pebble bed reactor core***

The reactor core of HTR-10 contains about 27 000 spherical fuel elements of 6 cm diameter, which form a pebble bed of 180 cm in diameter and 197 cm in average height. The mean power density of the core is 2 MW m^3.

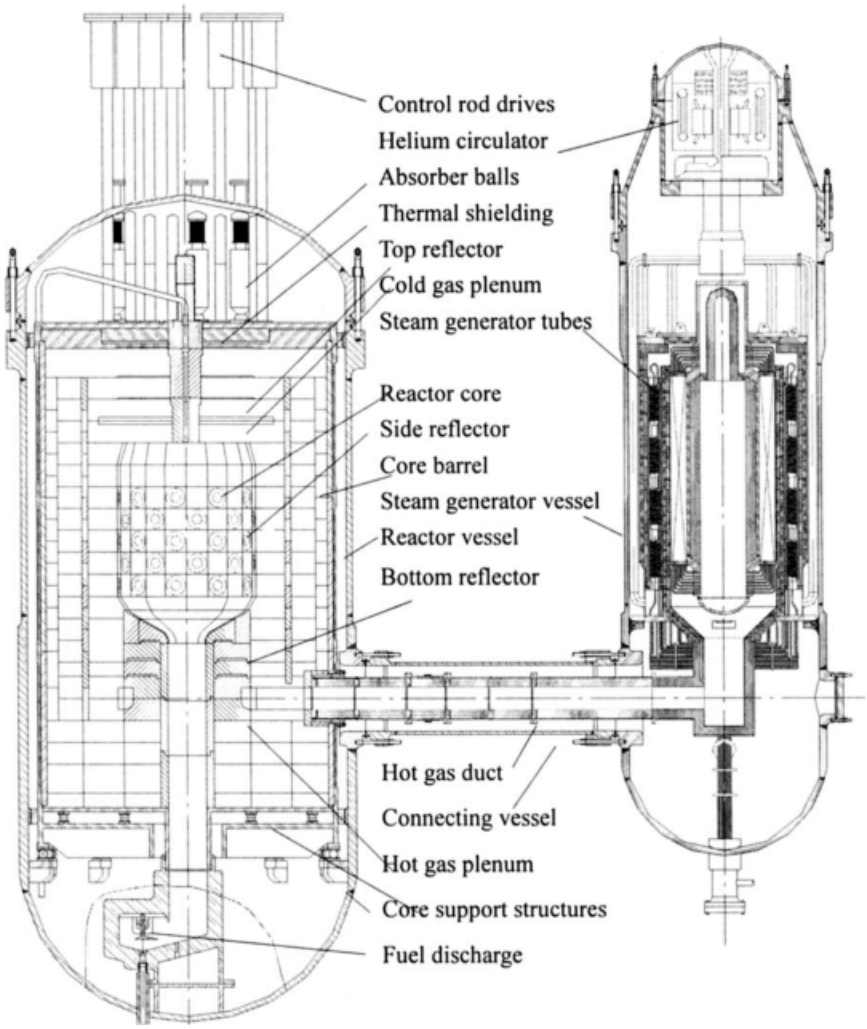
The fuel elements drop into the reactor core from the central fuel charging tube and pile up at the top of the core. The space of 40 cm height between the top of the core and the bottom of upper graphite reflector is a margin for accommodating the criticality calculation uncertainty. The fuel elements move downward in the reactor core and discharge through a tube of 50 cm inside diameter at the core bottom and a connected singularizer, after which the fuel elements are lined up and then pass the defective fuel separator and the burn-up measurement facility one by one.

The defective fuel elements and the scrap fragments should be sorted out and dropped into the scrap container. The fuel elements, which have not reached the burn-up target, will be re-circulated into the reactor core, and the spent fuel elements exceeding the burn-up target will be discharged and transported into the spent fuel storage tank, depending the burn-up measurement. New fuel elements are loaded into the core in a special charging facility in which the atmospheric isolation is realized to prevent air penetrating into the primary system and coolant being released into the environment. Transport of fuel elements in the fuel handling system is by means of pneumatic force and gravity. In order to reduce the power peaking factor of the HTR-10 core the fuel management utilizes a 'multi-pass' scheme. Each fuel element will pass through the reactor core five times in average before it reaches the burn-up target.

The HTR-10 uses helium gas as coolant, which has thermal and chemical stability, good compatibility with the core graphite material and metallic material of the primary system at high temperature condition; in addition, there is no phase transition. The cold helium at inlet temperature of 250 °C flows through the pebble bed core from the top to the bottom, and is heated up to 700 °C. The maximum temperature of the fuel elements appears in the bottom section of the reactor core under normal operating condition. The average burn-up of fuel elements in bottom section is higher than that of the whole reactor core. Therefore, both the helium gas and the fuel elements downwards flow help to reduce the maximum fuel temperature under normal operating conditions.

*1.2.3 **The side by side arrangement of the reactor pressure vessel and the steam generator vessel***

The pressure boundary of the HTR-10 primary circuit consists of the reactor pressure vessel, the steam generator pressure vessel and the hot gas duct vessel. The side by side arrangement of the reactor pressure vessel and steam generator vessel has the following advantages: (1) making the maintenance of the steam generator and the main helium circulator convenient; (2) reducing the probability of the core water ingress after rupture of steam generator tubes.



1.1 Layout of the HTR-10 reactor

The helium routing of the primary circuit is such as to insure the pressure boundary staying at low temperature. Therefore, all three steel pressure vessels (the reactor vessel, steam generator vessel and hot gas duct vessel) are in touch with the cold helium of about 250 °C from the main circulator. The cold helium enters the reactor core and flows through the pebble bed while it is heated up to 700 °C, then the hot helium flows through the hot gas duct and enters the steam generator. The heat is transferred to water in secondary circuit while the helium gas is cooled down to 250 °C. The cold helium enters the main circulator, and flows then along the inner wall of the steam generator vessel and flows into the outer coaxial pipe of the hot gas duct and enters the reactor pressure vessel where the partial cold helium is used to cool the wall of the reactor pressure vessel.

1.2.4 ***Passive removal of the decay heat***

The main heat transfer system is composed of the primary helium circulator, steam generator and feed pump, steam turbine, and circulating water system. During normal shut down operation the primary helium circulator runs and the core decay heat is transferred to the start up and shut down system through the steam generator, and then removed by the circulating water system.

In the accident of loss of coolant and depressurization the main heat transfer system would become ineffective, thus, no core cooling is foreseen at all, decay heat will dissipated via the core structure by means of heat conduction and radiation to the outside of the reactor pressure vessel, where a cavity cooler is installed on the wall of the concrete housing cavity. It is connected with the air coolers on the top of the reactor building.

This system works on the principle of the natural circulation of water and takes the decay heat via the air coolers to the atmosphere. In the HTR-10 two trains of passive decay heat removal system are installed and each of them has 100% capacity of decay heat removal.

The core diameter of HTR-10 is small and the average power density is low, therefore, for an accident of depressurization the maximum fuel temperature will be more or less that of the normal operation condition; this is due to the passive of decay heat removal. Even in the extreme condition of the failure of the two trains of the cavity cooler, the decay heat of the core could be dissipated through the wall of reactor concrete and the maximum temperature of fuel elements would be below the temperature limit of 1600 °C.

In fact, the cavity cooler of the reactor concrete housing cavity is also designed for cooling the vessel and its support; it will keep the temperature in the allowable range.
The monitoring of operation conditions and surveillance for cavity cooler are maintained to realize the necessary operation conditions.

*1.2.5 **Two independent reactor shutdown systems***

HTR-10 has two independent reactor shutdown systems: one is the control rod system and the other is the small absorber ball system. They are both placed in the side graphite reflector and are able to bring the reactor to cold shutdown conditions.

The control rod system consists of 10 control rods and their driving apparatus. The absorber material in the form of sintered B C rings is filled in the annular space between two coaxial tubes. On demand, the control rods are driven by the step motor and chain–chain wheel and can drop into channels of the side reflector by means of gravity.

The small absorber ball system is designed as the second reactor shutdown system, so-called the standby shutdown system. The basic principle is that when the rods do not drop, the small absorber ball system can be used for the reactor cold shutdown to keep it at sub-critical condition.
Owing to the strong negative temperature reactivity coefficients, the turn-off of the main helium circulator, which causes the reactor core to heat up, is sufficient to stop the chain reaction.

*1.2.6 **The barrier of confinement***

In any accidents of the HTR-10 the maximum temperature of the fuel elements could not exceed the temperature limit and a significant radioactivity release can be excluded. In addition, the low free uranium content of fuel elements, the retention of relative radioactivity by graphite matrix of fuel elements, and the negligible activated corrosion products in the primary coolant system will maintain the radioactivity of the primary coolant system at a very low level. In the depressurization accidents of the primary coolant, the impact of radioactivity release on the environment will be insignificant. Therefore, it is not necessary to provide containment for the HTR-10.

Therefore, a confinement without requirement of pressure-tightness is adopted. The concrete compartments, which house the reactor vessel, the steam generator vessel as well as other parts of primary pressure boundary are designed to be leak-tight, will function as the confinement. The confinement, together with the accident ventilation system, serves as the barrier against the release of radioactivity into the environment.

The functions of the confinement are defined as:

1. During normal operation conditions the exhaust system maintains the cavity of the concrete compartments at negative pressure to prevent the dissipation of radioactivity inside the cavity into the reactor building. Before emission via the chimney, the exhausted air is filtered to minimize the impact on the environment according to the ALARA principle;

2. In the depressurization accidents, after the pressure inside the cavity exceeds 0.1 bars above atmospheric pressure a repture disk in the exhaust pipe will be automatically opened.
3. The air inside the cavity will not be filtered but will be directly released into the atmosphere via the chimney.

The design of the confinement is based on a principle of low risk release. For the severe accident of depressurization the release of radioactivity is classified as prompt release and delayed release. In the prompt release the gaseous fission products in the helium coolant and solid fission products deposited on the graphite dust will be brought into the atmosphere in company with the release of primary coolant. In this case a small amount of radioactivity could released.

However, in the delayed release of the depressurization the additional fission products retained in the damaged coated particles and the graphite matrix of fuel elements, which are caused by the temperature increase in fuel elements, could be released. This release would last for several tens of hours. Therefore, the release through chimney without filtering at the early stage of depressurization would cause less impact on the environment.

1.3 Information from AGR tests on temperature limits of UCO TRISO fuel

1.3.1 AGR-1 irradiation data on the temperature limits of UCO TRISO under normal operation [13]

UCO TRISO is very robust; 300,000 particles were tested to burnups up to 19% FIMA without failure after 620 effective full power days. The AGR-1 irradiation resulted in fuel being exposed to very high temperatures for long times, well in excess of that expected in an actual HTGR/VHTR. For example:

- 15% of particle population saw temperatures in excess of 1250°C for 200 days
- 10% of particle population saw temperatures in excess of 1300°C for 100 days
- 5% of particle population saw temperatures in excess of 1350°C for 50 days
- 2% of particle population saw temperature in excess of 1400°C for 25 days

The AGR-1 95% confidence failure fraction is <1E-5, a factor of 20 better than the design in-service failure fraction of 2E-4. The more severe AGR-1 irradiation conditions compared to the vast majority of historic modular HTGR designs suggest substantial fuel performance margin. Currently there is no known normal operation temperature limit for UCO TRISO. We are awaiting results from the 1400°C irradiation capsule in the AGR-2 experiment.

Two to three particles (2.0 to 2.6E-05 at 95% confidence) had failed SiC after irradiation. PIE indicates buffer cracking led to IPyC and then SiC cracking. Partial cracking of SiC was found in other particles. Cesium release into the matrix was very low unless there is a failed or defective SiC layer. No significant Pd corrosion found. Although 1% of Pd was found outside of SiC layer, little to no cesium release was measured outside of the SiC layer.

In terms of model predictions, Ag release under irradiation is reasonably well predicted and Cs release from intact particles overall is conservatively well predicted. Sr is largely over-predicted on two compacts and generally within factor of 5 for most other compacts. Cs and Sr comparisons globally agree within the 10× range; Ag was much closer. New diffusion coefficients are needed for Eu and Sr

[13] What does AGR-1 tell us About Temperature Limits of UCO TRISO Under Accidents? David Petti, Bob Morris, Paul Demkowicz, John Hunn, Blaise Collin, Richard Hobbins Oak Ridge Laboratory 2013 in: https://www.iaea.org/NuclearPower/Downloadable/Meetings/2013/2013-06-10-06-12-TM-NPTD/14_uco_limits_and_fission_product_behavior_accident.pdf

in UCO because of different chemical form than in UO2 (carbide vs. oxide) and the greater mobility of Eu and Sr carbides.

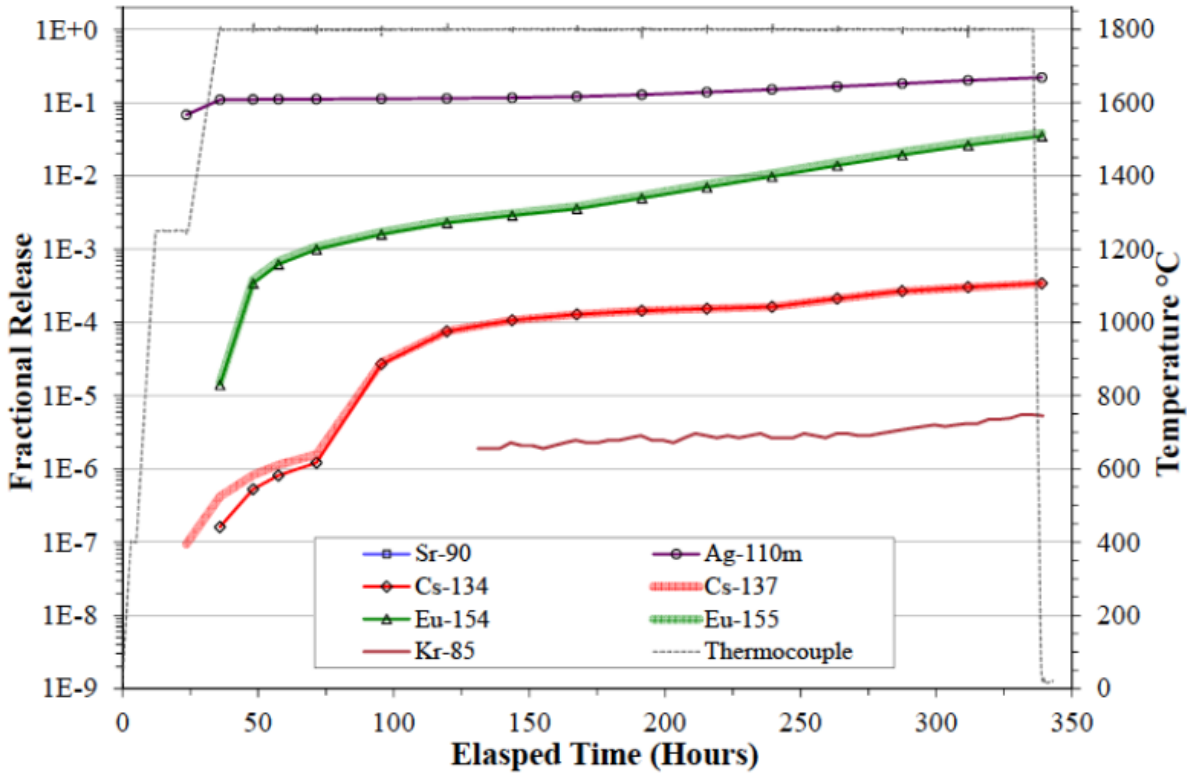


Fig. 1.2 Fractional releases of fission products in function of time in the temperature of 1800 °C

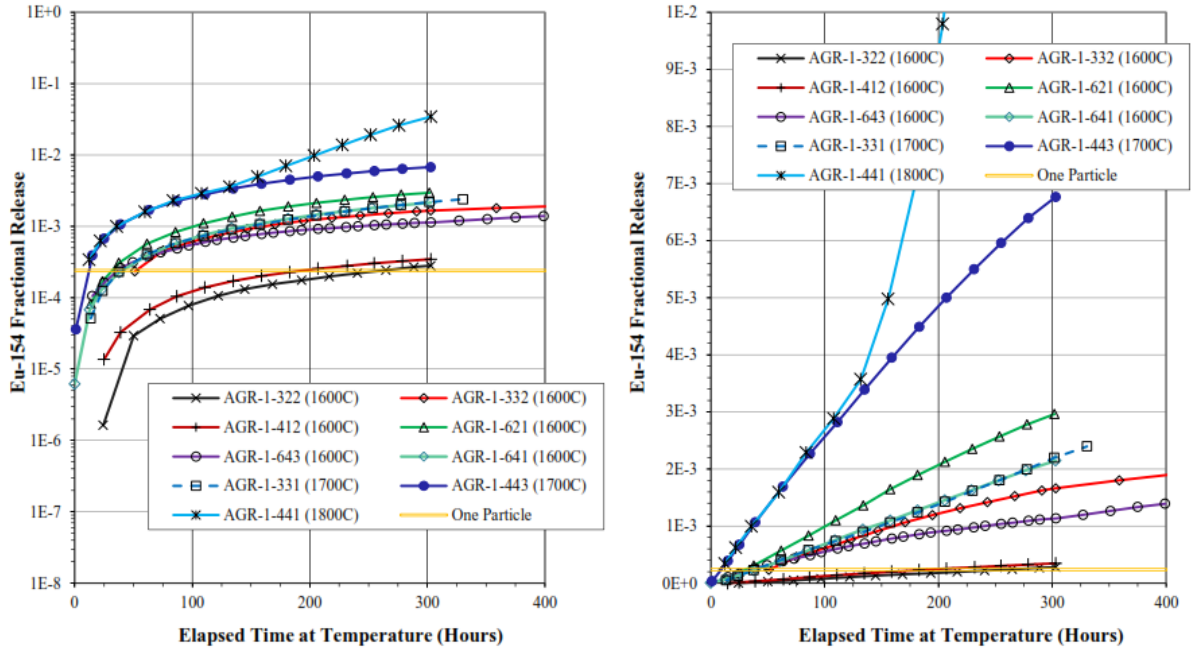


Fig. 1.3 Fractional releases of Eu-154 in high temperatures (1600 – 1800 °C) measured in AGR-1

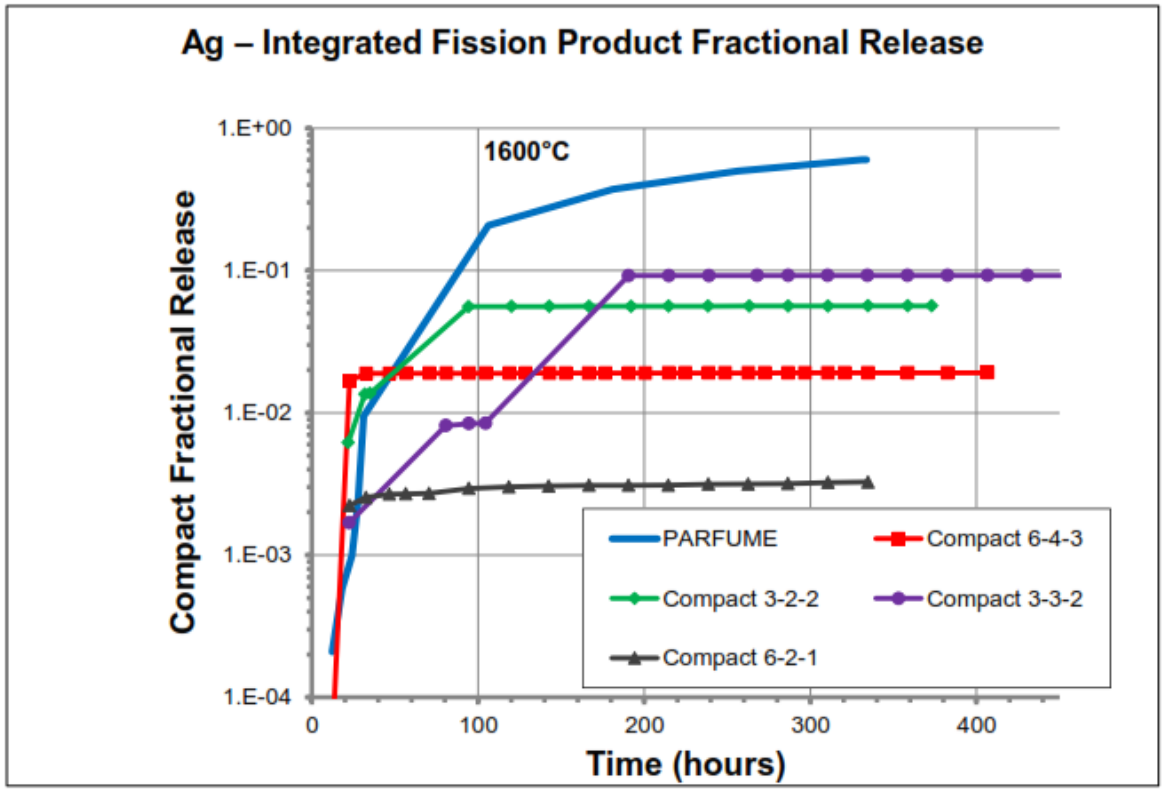


Fig. 1.3 Fractional releases of Ag in high temperatures (1600 – 1800 °C) measured in PARFUME and Compact experiments

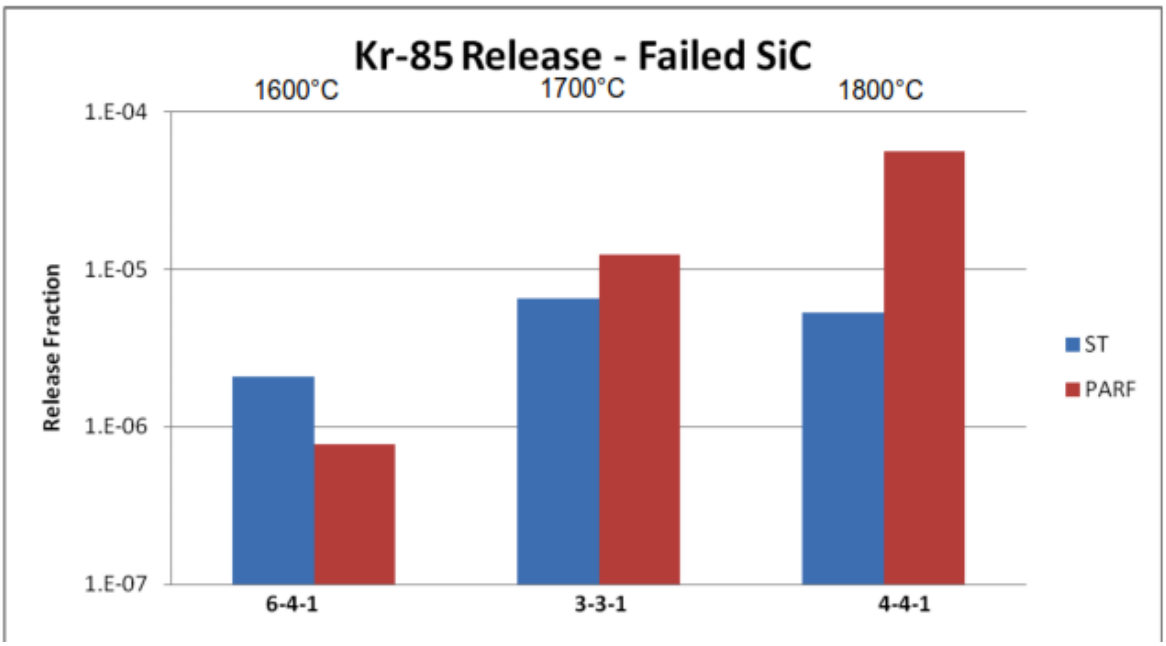


Fig. 1.4 Fractional releases of Kr-85 in high temperatures (1600 – 1800 °C) measured in ST and PARFUME experiments

*1.3.2 **Conclusions***

- Accident performance of UCO TRISO is very robust (300 hours at 1600, 1700 and 1800°C). Releases are very low
- No full particles failures observed in testing to date (9 compacts = ~36,000 particles). This implies a 95% failure fraction of ~ 8E-05, a factor of 7.5 margin relative to the prismatic reactor specification. We expect the margin to be 10× when all testing is complete
- Recall that only 2-3% of reactor core sees 1600°C in conduction cooldown scenario
- No release from intact particles. Release is coming from material that diffused into the matrix under irradiation
- No SiC degradation noted at 1800°C as was seen in German UO TRISO – Releases imply different physics
- Cesium releases above the level of one particle are related to SiC failures that occurred during heating

Model calculations using IAEA TECDOC diffusion coefficients significantly overpredict releases from intact particle

- SiC diffusivity too high
  - Ag › 1 to 3 orders of magnitude
  - Cs › 2.5 to 3.5 orders of magnitude
  - Sr › 1.5 to 3.5 orders of magnitude
  - For particles with failed SiC…
- Cs and Kr releases well reproduced (w/out “intact” contribution)
  - Cs diffusion coefficient in kernel is too high
  - very low contribution from intact particles
- Sr under-predicted by factor of 10 kernel diffusivity too low (UO used)
- Need to back out Eu diffusion coefficients from this data

2 Safety related systems of HTGR plant[14]

2.1 General assumptions for the design

The use of helium as a coolant, which is both chemically and radiologically inert, combined with the high temperature integrity of the fuel and structural graphite, allows the use of high primary coolant temperatures (800 to 900°C) which yield high thermal efficiencies. With these high temperatures, the use of a closed cycle gas turbine is justified. This increased the efficiency of HTGR over a steam plant (from ~35% to ~45%), thus reducing the unit capital cost. It also removes external sources of contamination of the nuclear circuit, as there is no system with a higher pressure than the helium. Without the possibility of leakage into the helium circuit the need for on-line clean up systems is largely reduced.[15]

However, the pebble bed reactor planned for Poland will work with a gas-steam cycle and will be provided with a steam generator side-by-side with the reactor pressure vessel.

The Standard MHTGR plant consists of four reactor modules and two turbine generator sets to achieve the nominal 558-MW(e) plant rating. Each module consists of separate vertical reactor and

[14] DOE-HTGR-86-011 Revision 3 Volume 1 Probabilistic risk assessment for the standard modular high temperature gas-cooled reactor

[15] IAEA TECDOC 1198 Current status and future development of modular high temperature gas cooled reactor technology, 2001.

steam generator vessels connected by a horizontal coaxial cross duct. The major components of the nuclear steam supply system (NSSS) portion of the plant are contained within the MHTGR as shown in Fig. 2-1. The design parameters for the NSSS are listed in Table 2-1.

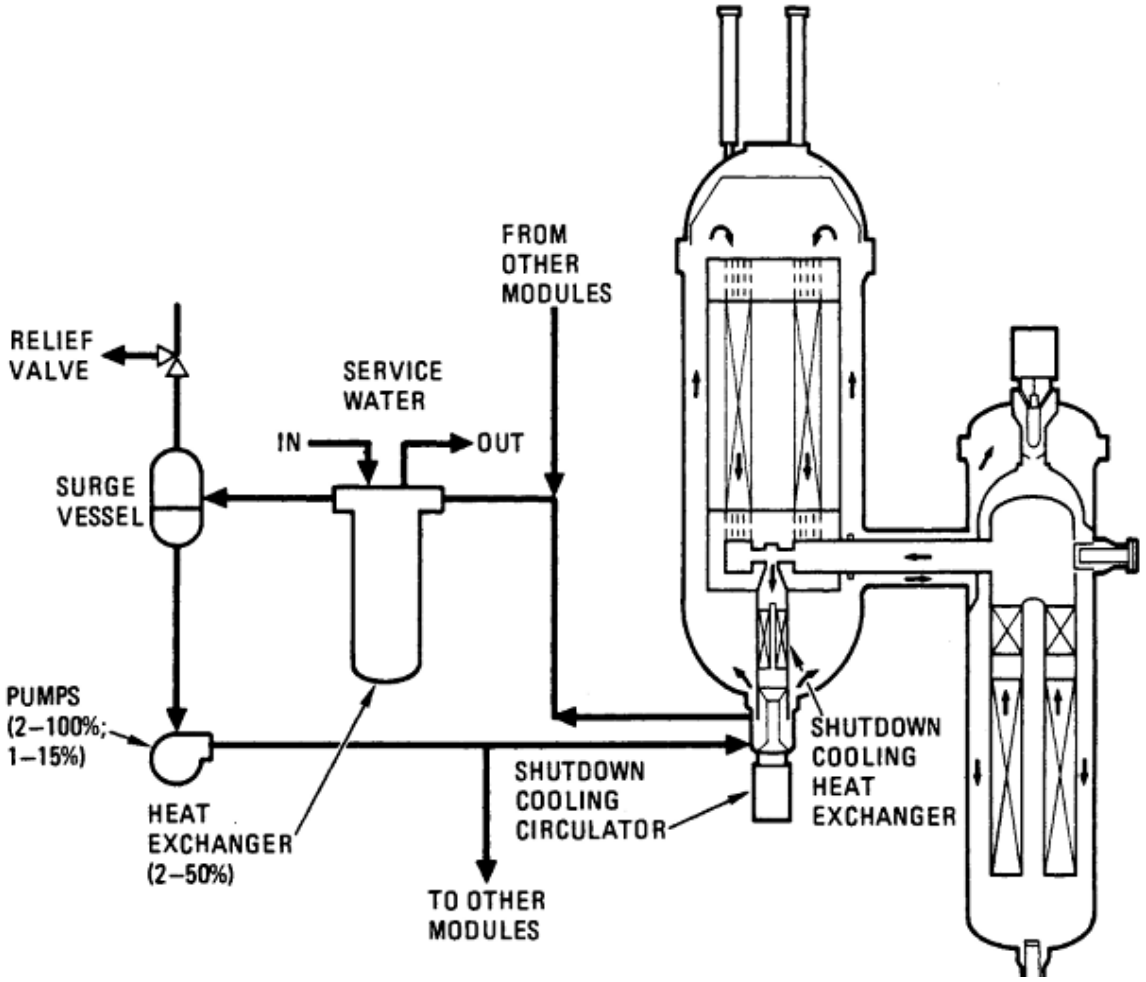


Fig. 2.1 Shutdown cooling system arrangement[16]

[16] DOE-HTGR-86-011 Revision 3 Volume 1 PROBABILISTIC RISK ASSESSMENT FOR THE STANDARD MODULAR HIGH TEMPERATURE GAS-COOLED REACTOR

NSSS DESIGN PARAMETERS

Item	Parameter
Reactor System	
Modules per station	4
Power per module, MW(t)/MW(e)	350/140 nominal
Coolant (helium) pressure at rated power	Helium at 6.38 MPa (925 psia) at circulator discharge
Cold helium temperature (at circulator discharge)	258°C (497°F)
Hot helium temperature (at core exit)	687°C (1268°F)
Feedwater temperature/pressure	193°C/20.68 MPa (380°F/3000 psia)
Steam temperature/pressure	541°C/17.3 MPa (1005°F/2515 psia)
Configuration description	Side-by-side (SBS)
Vessel material	Carbon steel - Mn-Mo, SA 533, Grade B, Class 1
Reactor vessel overall height, with CRDS and shutdown circulator	28.9 m (94.8 ft)
Reactor vessel outside diameter	6.8 m (22.4 ft)
Number of Components Per Module	
Steam generators	1
Circulators	1 main, submerged electric motor-driven
	1 shutdown cooling, electric motor-driven
Shutdown heat exchangers	1
Control rods	30 (6 inner, 24 outer reflector rods)
Reserve shutdown channels	12 (inner row of core fuel elements)
Start-up system (flash tank)	1
Core and Fuel Cycle	
Fuel element configuration	Prismatic hex-block, 20.78 cm (8.18 in.) sides x 79.3 cm (31.22 in.) height
Fissile material	UCO
Power density	5.91 W/cm^3 (330.43 Btu/h-in.^{3})
Power peak/average axial ratio	1.4:1
Average enrichment	19.9% U-235
Fertile material	ThO_2

2.2 Reactor core subsystem

The reactor core subsystem consists of fuel elements, hexagonal graphite reflector elements, plenum elements, startup sources, and reactivity control material, all located inside the reactor pressure vessel. The reactor core, together with graphite components of the reactor internals subsystem, constitutes a graphite assembly which is supported on a graphite support structure and restrained by a core lateral restraint structure.

The standard hexagonal fuel elements are stacked in columns that form an active core annulus. Columns of hexagonal graphite reflector elements are in the central region and surround the active core. The core produces 350 MW(t) of power at a power density of 5.9 MW/m3~

Placed on top of the upper graphite reflector are plenum elements, one per column, for channeling the coolant flow. These plenum elements contain radiation shielding material. Beneath the active core are hexagonal graphite reflector elements. These lower reflector elements continue the coolant hole pattern from the active core. Flow in these channels exits into the core support blocks. The reactor core subsystem contains both fixed and movable poison for normal operation. The fixed poison is in the form of lumped burnable poison rods and the movable poison is in the form of metal clad control rods. In the event that the control rods become inoperable, backup reserve shutdown control is provided in the form of boronated pellets that may be released into the core.

The functions of the reactor core subsystem are to generate heat from fission energy produced in a well controlled self-sustaining neutron chain reaction, and to transfer the heat to the helium primary coolant flowing through the core.

The reactor core subsystem also performs the function of retaining radionuc1ides in the fuel with fuel particle coatings upon which the safety of the MHTGR is based. As a part of accomplishing this, the annular geometry of the core functions to promote passive heat removal via conduction and radiation while other features function to control the effect of chemical attack and control the generation of heat under off-normal conditions.

MHTGR fuel particles consist of both fertile (Th02) and fissile (UCO) material. Both fuel types are in the form of dense microspheres coated with a TRISO coating whose primary purpose is to retain fission products. The fuel kernel and particle coating layers provide resistance to gaseous and metallic fission product release. The coated particles are blended and bonded together with a carbonaceous binder into the form of fuel rods. The rods are then used to construct the fuel elements. The bonding of fuel particles into rods provides an additional barrier to metallic fission product release through graphite adsorption mechanisms.

2.3 Reactor internals subsystem

The reactor internals subsystem consists of the core lateral restraint, permanent side reflector, graphite core support structure, metallic core support structure, upper plenum thermal protection structure, and the hot duct.

The core lateral restraint and the permanent side reflector surround the core; the graphite core support structure and metallic core support structure are located below the core; the upper plenum thermal protection structure is located above the core; and the hot duct is located within the cross duct between the reactor vessel and the steam generator vessel.

The principal function of the reactor internals subsystem is to provide support and lateral restraint for the reactor core. Other important functions are to channel the primary coolant flow to the core, to control the amount of core coolant bypass flow, and to mix the core exit coolant flow. The reactor internals also augment shielding of the reactor vessel from core radiation. Furthermore, the core lateral restraint, permanent side reflector, graphite core support structure, and metallic core support structure remove core heat and assist in controlling heat generation. These functions are performed by maintaining cooling pathways and the geometry necessary for reactivity control material movement.

2.4 Neutron control subsystem

The neutron control subsystem consists of the drive mechanisms for positioning the control rods, the rod controls, the reserve shutdown control equipment (RSCE) with its controls, and the instruments for measuring neutron flux levels within the reactor vessel (i.e., in-vessel flux mapping units and startup detectors) and around the perimeter of the reactor outside the vessel (i.e., ex-vessel flux detectors). Most of this equipment is configured into assemblies which are normally installed in penetrations in the top or bottom of the reactor vessel.

These assemblies are periodically removed either to provide access to the core for refuelling or for maintenance of the equipment.

Five types of assemblies are provided for each reactor module:

1. Twelve outer neutron control assemblies.
2. Six inner neutron control assemblies.
3. Six ex-vessel neutron detector assemblies.
4. Three startup detector assemblies.
5. Five in-vessel flux mapping units.

2.5 Vessels and duct subsystem

The vessels and duct subsystem for each module consists of a reactor vessel and a steam generator vessel placed side-by-side and connected by a cross duct. The vessels and duct subsystem includes the vessel penetrations, closures, and thermal insulation and the main steam and feedwater isolation valves. The principal functions of the subsystem are to contain the primary coolant inventory and to provide a coolant flow path for transfer and transport of thermal energy. In addition, the subsystem provides support for the reactor core and internals, for the steam generator and main circulator, and for the shutdown cooling heat exchanger and circulator. The vessels and duct subsystem is located below grade level and is enclosed in a concrete structure. The steam generator vessel is located at a somewhat lower elevation than the reactor vessel. This allows the steam generator to be thermally protected and isolated following a loss of forced circulation. The vessels and duct subsystem is bottom-supported by the reactor cavity through attachments anchored to the vessels at or below the level of the cross duct. The cross duct is supported through its connections to the vessels.

2.6 Reactor building subsystem

The reactor building is a multi-cell, reinforced concrete structure housing the nuclear steam supply system (NSSS) and ancillary systems and components. Set below grade, the reactor building is configured as a 18.3-m inside diameter, vertically oriented right cylinder topped by a rectangular prism. Minor portions of the building extend above grade, principally' the reactor cavity cooling system (RCCS) intake and exhaust structure, and the main steam and feedwater isolation valve enclosure. Four such buildings are arranged in a row and are served by a bridge crane running their entire length. A steel-framed maintenance enclosure shelters the entire crane service area.

In general, the reactor building's functions and associated requirements are dictated by the needs and characteristics of the equipment it houses. However, the reactor building also serves to protect that equipment from external hazards and contributes to the capability to control both normal and post-accident onsite radiation levels.

The cylindrical portion of the below-grade cavity is subdivided into a number of vertical cells which house NSSS equipment and provide access. Those cells housing the reactor and steam generator occupy the major portion of the building. The steam generator and reactor are separated by a 1.S-m (S-ft)-thick wall which is penetrated by the crossduct. Other cells provide personnel and equipment access and pipe and cable ways.

The rectangular portion of the building is divided into two levels and subdivided into several compartments. The majority of this area is occupied by RCCS ducting and the cavity vent path. Other spaces house helium purification system (HPS) equipment, PPIS equipment, and other nuclear auxiliaries which are dedicated to each reactor.

The reactor building serves as an enclosure that can be vented in a controlled manner providing an additional attenuating barrier to radionuclide releases. The vent paths from the reactor cavity to the steam generator cavity and from the steam generator cavity to the atmosphere, as well as the RCCS ducts, follow tortuous routes to limit neutron streaming from the reactor building. To maintain the requisite reactor cavity environmental conditions while limiting heat load on the steam generator cavity chiller, the reactor cavity vent path is fitted with blowout panels. Similarly, to maintain control of the steam generator cavity environment, the steam generator cavity vent path is fitted with gravity dampers, which open on overpressure and then reshut. These dampers are located below elevation -7m to provide them protection from external hazards, with ultimate release of steam or helium to the atmosphere made through louvered openings located above elevation 3.6 m adjacent to the main steam isolation and relief valve enclosure.

The entire reactor building is designed structurally to withstand the requisite levels of intensity for external and internal hazards to ensure that the equipment it houses can function as required to meet the investment protection and public health and safety criteria. This includes the structural framework for the maintenance enclosure, which is to be designed not to collapse under design basis conditions.

2.7 Heat transport system

The heat transport system (HTS) consists of the steam generator subsystem and the main circulator subsystem.

The steam generator subsystem serves to limit the release of radionuclides by maintaining its primary/secondary coolant pressure boundary integrity, thereby containing radionuclides (contaminated primary coolant), controlling radionuclide transport (tritium diffusion) from the primary coolant, and preventing chemical attack.

2.8 Shutdown cooling system

The shutdown cooling system (SCS) consists of the shutdown cooling circulator subsystem, the shutdown cooling heat exchanger subsystem, and the shutdown cooling water subsystem (SeWS).

The principal function of the SCS is to provide a second means of forced circulation residual heat removal from the shutdown reactor by transferring this heat to the service water subsystem when the HTS is unavailable. In addition, portions of the SCS have the function of containing the primary coolant inventory.

2.9 Reactor cavity cooling system

The RCCS is required to remove heat from the reactor cavity during normal power producing operations in order to protect the concrete structures from overheating. When the reactor is shut down, decay heat is normally removed from the vessel through the steam generators to the main condenser, or by the SCS. However, in the event these paths are not available, decay heat is removed from the core and vessel by conduction and radiation. During such passive cooling, the vessel side walls and concrete structure temperatures are limited to acceptable values by the RCCS. Under these conditions decay heat .is radiated from the vessel wall to the air cooled RCCS cooling panels, and then transported to the atmosphere by natural convection air flow.

Since the RCCS must remove both normal and decay heat loads, the system must function continuously while the reactor is at power or generating significant decay heat. The ReeS is a completely passive, air-cooled system which provides a high degree of reliability. The system removes heat from the reactor cavity by the natural convection of outside air through the cooling panels located in the reactor cavity. The cooling panels are divided into four quadrants, each quadrant having an

annular inlet air duct and rectangular outlet duct routed inside the inlet passage. This arrangement protects the structural concrete from the hot outlet air.

2.10 Steam and water dump subsystem

The steam and water dump subsystem serves to further limit ingress of water into the primary coolant as a result of a steam generator tube leak or rupture. This is accomplished by dumping the steam/water inventory of the steam generator into the subsystem's dump tank following isolation of the steam generator from the feedwater and steam headers. The subsystem dump action minimizes possible damage to the reactor core by limiting the amount of water made available for fuel hydrolysis and graphite oxidation.

2.11 Pressure relief subsystem

The pressure relief subsystem prevents the vessel system from exceeding its design pressure, hence providing overpressure protection for the primary coolant pressure boundary. The subsystem is composed of two identical pressure relief trains interlocked so that at least one is available at all times. Both trains are connected to the steam generator vessel upper head at the main circulator discharge where the primary coolant pressure is nominally the highest. Each train consists of a pilot-actuated, spring-loaded safety relief valve in series with a rupture disk, both of which must be activated to achieve pressure relief. The effluent is discharged to the steam generator cavity.

2.12 Helium purification subsystem

The helium purification subsystem (HPS) for each reactor module consists of a specific sequence of gas processing components, plus related piping, valves, controls, and instrumentation. This system purifies a helium side stream from the primary coolant system at a rate of 386.4 kg/h (850 lb/h) for each reactor module during normal full power operation. The HPS removes chemical impurities in order to maintain their concentration in the primary coolant helium within prescribed limits and removes the gas borne activity contained in the side stream flow. Excess flow is returned to the primary system or transferred to storage as appropriate.

The HPS provides purified helium to equipment such as the main circulator, shutdown circulator, the pressure relief equipment on the reactor vessel, and valve penetrations in the vessel. A simplified BPS flow schematic for each reactor module is shown in Fig. 4-27 along with an indication of the primary function for each principal component in a helium purification train. Also shown in the figure is the separate regeneration train used to regenerate the dryers and low-temperature adsorbers in the BPS. One regeneration train services two reactor modules.

During depressurization of the reactor vessel for refueling and maintenance operations, the HPS purifies the discharged helium at no more than four times the normal flow rate. Following depressurization, the BPS maintains the reactor vessel pressure slightly below atmospheric pressure.

Should a small primary coolant leak be detected by a drop in primary coolant pressure, the HPS is used to depressurize the reactor vessel automatically, in order to minimize the effect of the leak. The HPS is actuated manually by the operator in the event forced convection cooling and the RCCS are unavailable to cool the reactor core. This action is taken to prevent overstress conditions that would exist if the vessel were pressurized.

3 Pebble bed high temperature reactor fuel

3.1.1 TRISO Fuel geometry

In the presentation of Petti[17] a comparison of three types of TRISO fuel is provided.

German UO2 TRISO small compacts

HFR-P4 higher burnup and fluence than AVR and much of HFR-K series of pebbles – 12-15% FIMA, 5-8 x 10^{25} n/m^2

- 500 micron UO2– 36 micron SiC (except P4-2 which had 51 micron SiC)
- 1630 particles/compact
- Heated at 1600°C

- **Japanese UO2**

TRISO irradiated in HRB22 test at ORNL at ~1100°C for 89 EFPD to burnup of 4.8% FIMA and fluence of 2.1x10^{25} n/m2 (highlyaccelerated irradiation)

- ACT3 – heated 25 particles to 1700°C for 270 hrs
- ACT4 – heated 25 particles to 1800°C for 222 hrs; one failed particle

- **US UCO TRISO AGR-1 compacts**

- 14-19% FIMA, 3-5 x 1025 n/m
- 350 micron UCO
- 4150 particles/compact
- Heated at 1600, 1700 and 1800°C

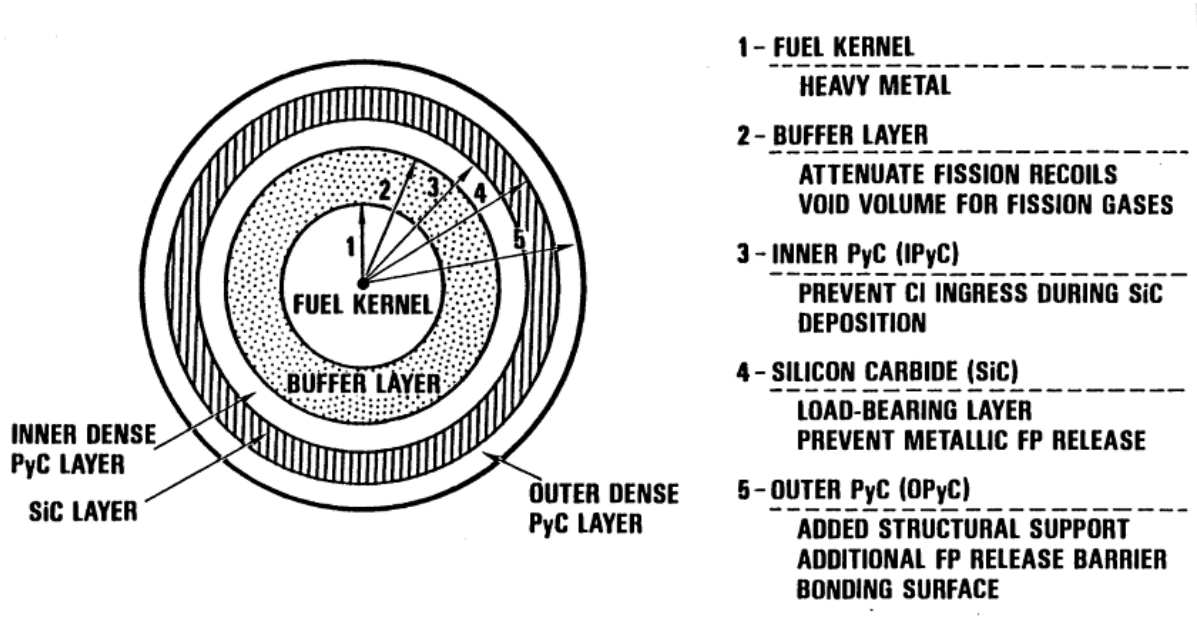


Fig. 3.1 TRISO - coated fuel description

[17] D. Petti: Comparison of German,Japanese and US Heating Test Data June 10-13, 2014 , IAEA. Vienna, Austria

3.2 Experimental studies of fission product releases from TRISO fuel ,

3.2.1 German TRISO Fuel Performance Envelope and Limits - Normal Operations and Accident Conditions[18]

The German High Temperature Reactor Fuel Development Program successfully developed, licensed and manufactured many thousands of spherical fuel elements to power the experimental and prototype High Temperature Reactors (HTR). In the 1970s, this program extended the performance envelope of HTR fuels by developing and qualifying the high-enriched (Th,U)O2 TRISO-coated particle system. In the 1980s, a program direction change was made to a low enriched UO2 TRISO-coated particle system. This change was coupled with high-quality fuel manufacturing specifications designed to meet new HTR plant requirements for inherent safety under normal operation and accident conditions.

This program met and exceeded these challenges by manufacturing, in production scale facilities in the late 1970s and early 1980s, near defect free HEU (Th,U)O2 TRISO- and the LEU UO2 TRISO coated fuel particles, homogeneously distributed within a graphite matrix with very low levels of uranium contamination. The performance of these fuels was successfully demonstrated in reference 60 mm-diameter elements in three primary areas of development; manufacturing, irradiation testing under normal operating conditions, and accident simulation testing. In terms of demonstrated performance for advanced HTR applications, the experimental failure statistics are: (i) from manufacture [~4.7x10-5 - LEU UO2 , ~2.2x10-5 – HEU (Th,U)O2] significantly lower than the HTR Modul requirement of 6x10-5; (ii) from irradiation under normal operating conditions [4.5x10-5 – LEU UO2, ~4.4x10-5 – HEU (Th,U)O2] >2X lower that the HTR Modul requirement of 1x10-4; and (iii) under accident conditions [5.1x10-5 – LEU UO2, 4.8x10-5 – HEU (Th,U)O2] ~8X lower than the HTR Module requirement of 4x10-4.

Inherent safety features of modern small modular HTRs are, in part, provided by the high level of fuel quality of TRISO coated fuel particles. The work conducted within the German Fuel Development Program over several decades constitutes a convincing demonstration of fuel excellence in manufacture, continuing through irradiation and accident simulation testing. The low levels of particle defects produced during manufacture were not significantly altered by extensive accelerated and real-time irradiation testing, or through accident simulation testing under dry and wet environments.

3.2.2 Studies in Kharkov Institute of Physics and Technology

The studies of structural materials and nuclear fuel for high-temperature gas cooled reactors VGR-50, VGM and VG-400 performed at National Science Center "Kharkov Institute of Physics and Technology" covered three types of fuel[19], namely TRISO UCN carbonitride fuel of PyC-SiC-PyC design and (PyC+SiC)-SiC-(PyC+SiC) design, (Th,U)O2 fuel and UO2 (SiO2, Al2O) fuel.

The tests carried out for long-duration operation efficiency under irradiation of carbonitride fuel at a temperature 1500 °C up to a burn-up of 18 % fima (fissions per initial metal atom) have shown a high working efficiency of the developed coated particle (CP) fuel (R/B no more than $6.0 \cdot 10^{-6}$ for the PyC-SiC-PyC design and $3.5 \cdot 10^{-6}$ for the (PyC+SiC)-SiC-(PyC+SiC) design) that is more than twice higher than the required planned values for a burn-up.

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[18] IAEA Meeting Report (I3-TM-45373) Technical Meeting on the Re-evaluation of Maximum Operating Temperatures and Accident Conditions for High Temperature Reactor (HTR) Fuel and Structural Materials IAEA Headquarters 10–12 June 2013

[19] Dr. Mykola Odeychuk: The Results Of Fuel Behavior Studies And HTGR Core Components Under High Temperature Conditions Technical Meeting on High-Temperature Qualification of High Temperature Gas Cooled Reactor Materials, June 10-13, 2014 , IAEA. Vienna, Austria

(Th,U)O fuel. The in-reactor service-life tests were performed on the mixed oxide uranium-thorium fuel. At irradiation temperature 1600 °C this fuel was operated successfully up to the burn-up of 13.4 % fima.

UO2 (SiO2, Al2O) fuel. The radiation resistance of microspherical fuel based on uranium dioxide with doped alumosilicate additives in free backfill condition was investigated at a temperature of 1250 C up to a burn-up of 2,5 % fima (fast neutron fluence $1{,}0 \cdot 10^{20}$ cm^{-2}).

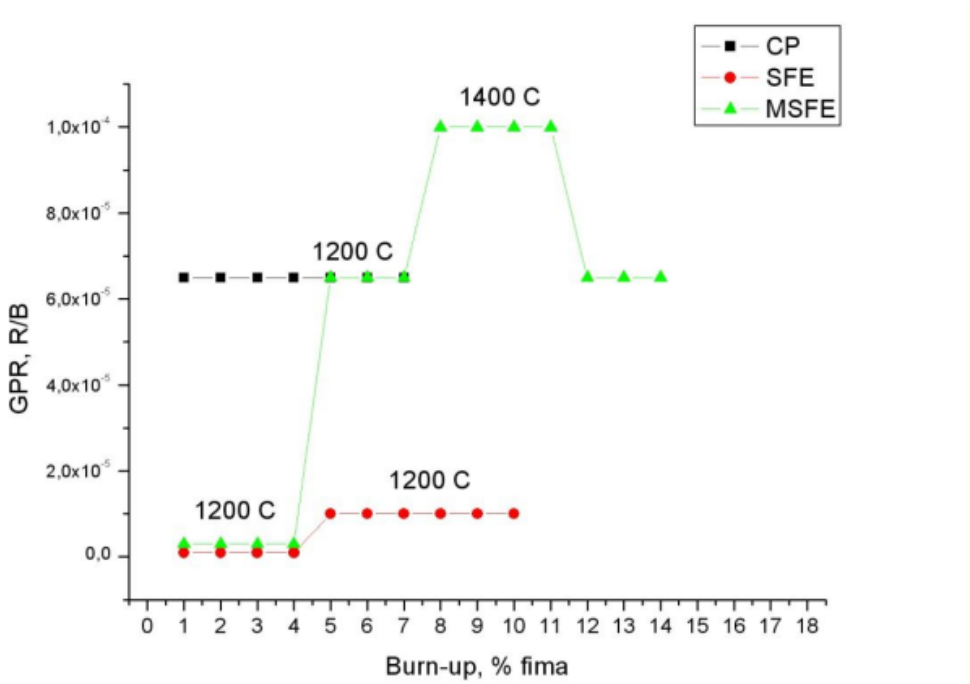

Fig. 3.2 Kr-85 release from SFE (spherical fuel elements), MSFE (Microspherical fuel elements) and 1-type CP (coated particle)

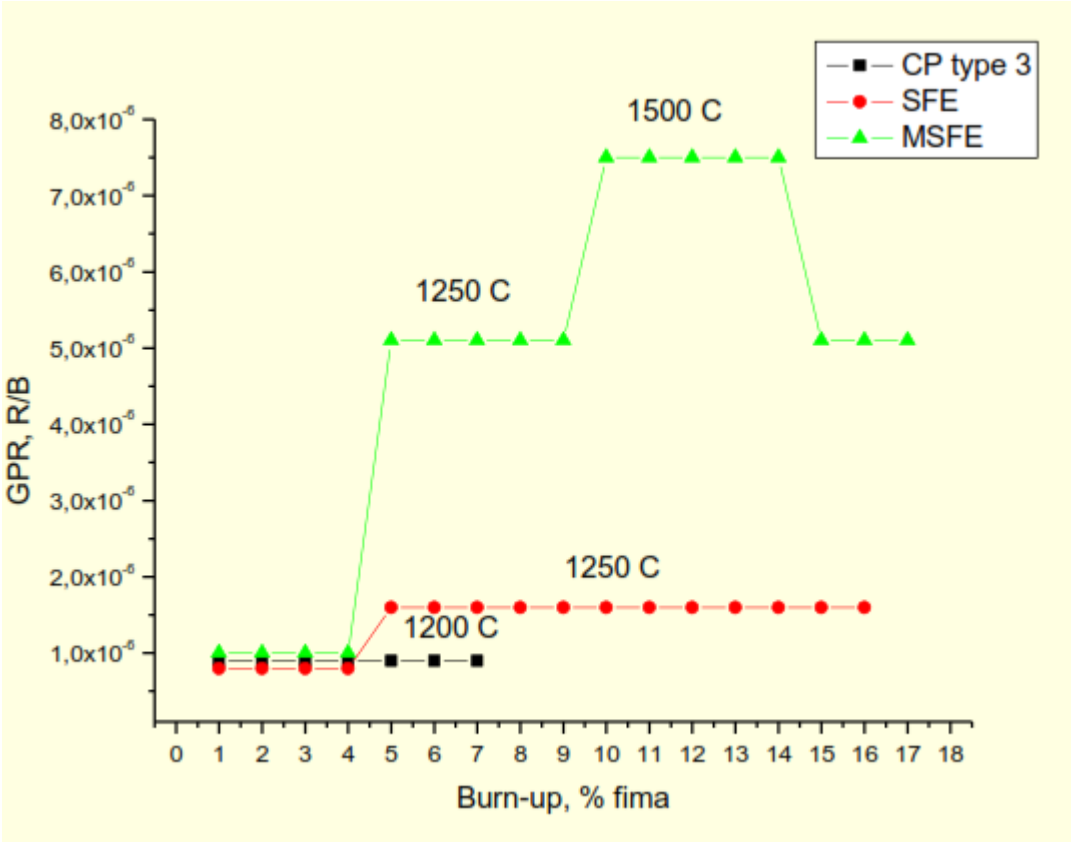

Fig. 3.3 Kr-85 release from SFE, MSFE and 3-type CP

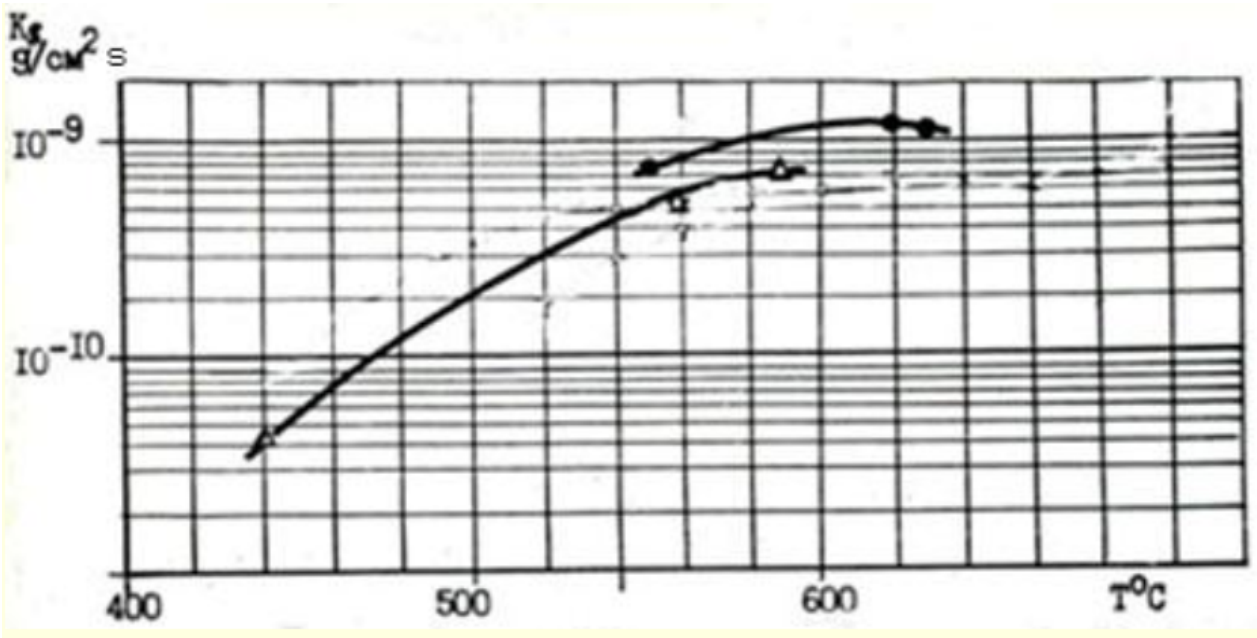

Fig. 3.4 Dependence of oxidation rate of graphite from surface temperature of SFE:

^ - GSP SFE; ● - standard SFE

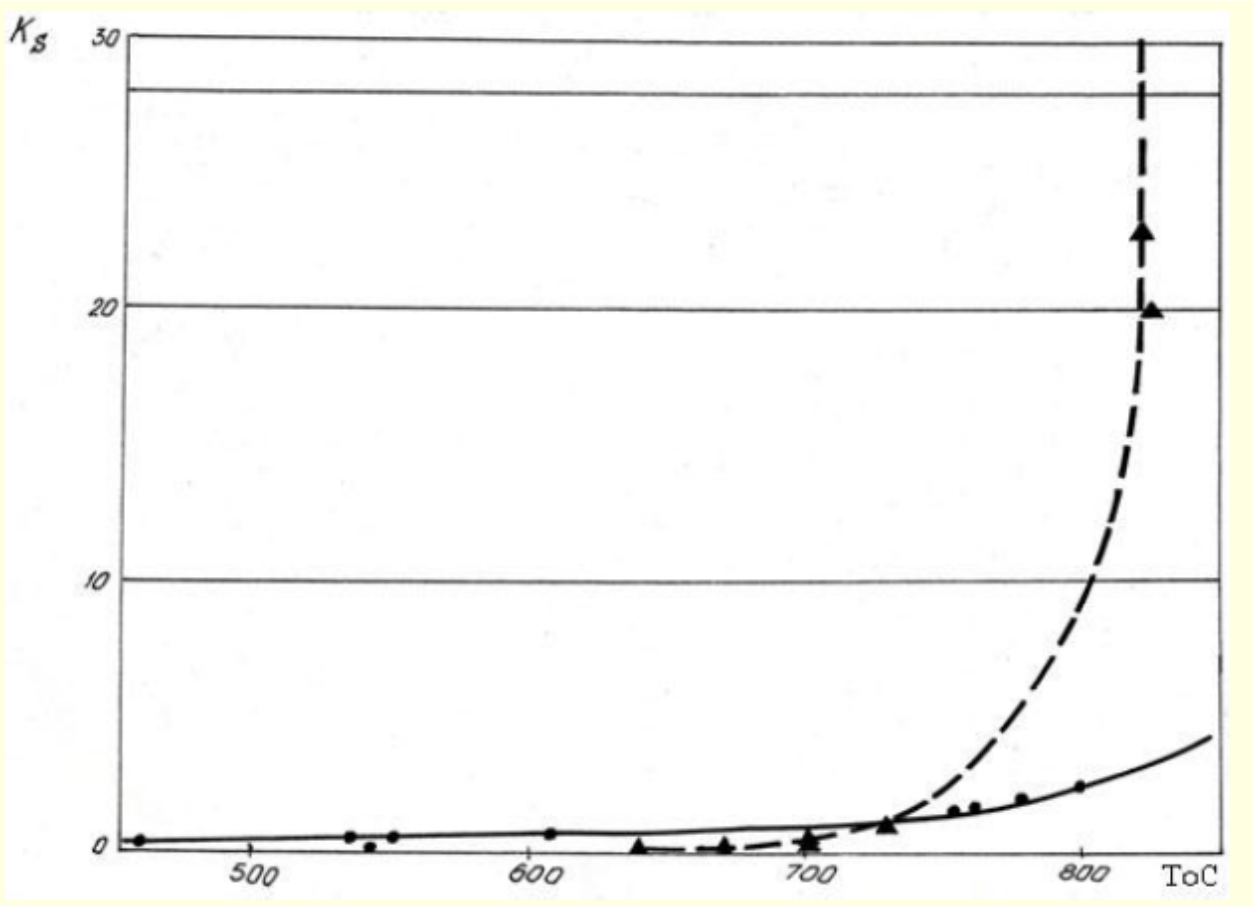

Fig. 3.5 Dependence of oxidation rate of graphite from surface temperature of SFE:

^ - standard SFE; ● - GSP SFE

3.2.3 ***Conclusions***

1. The results of high-temperature heat tests showed that the working capacity of coated particles is conserved up to 1800 C. Above 1900 ^{o}C the degradation of properties of coating of silicon carbide is begining, and after 2100 °C – a catastrophic destruction occurs.

2. Microspherical fuels developed in NSC KIPT composed of coated particles and full-scale spherical fuel elements conserve operational properties under normal conditions of HTGR operation at 1500 °C up to 20 % fima, at 1600 oC - to 15 % fima at GPF rate up to 10^{-6}.

3. Under neutron irradiation GSP graphite and CCCM materials, manufactured by the gas-phase technology, undergo isotropic shrinkage slightly dependent on the irradiation temperature and neutron fluence (above $3 \cdot 10^{21}$ n/cm^2).

4. GSP graphite did not reveal any degradation of properties up to the investigated neutron fluence of 1.10^{22} cm^{-2}

5. At impurity level of oxygen (1-2) vpm[20] in the helium coolant and surface temperature of spherical fuel elements up to 700 ^{o}C, long operation of spherical fuel elements under HTGR operating conditions is possible.

[20] Vpm – volume per million

6. At impurity level of oxygen (1-2) vpm and above in the helium coolant and surface temperature of spherical fuel elements above 800 ^{o}C there is a significant degradation under HTGR operating conditions.

7. For prevention of corrosion of spherical fuel elements in oxygen-containing helium coolant medium is necessary to develop anticorrosive coatings of spherical fuel element surface.

*3.2.4 **Melt wire temperature measurements in AVR fuel***

The paper of Nabielek , Verfondern, and Kania, presented the results of melt wire temperature measurements in AVR fuel[21].

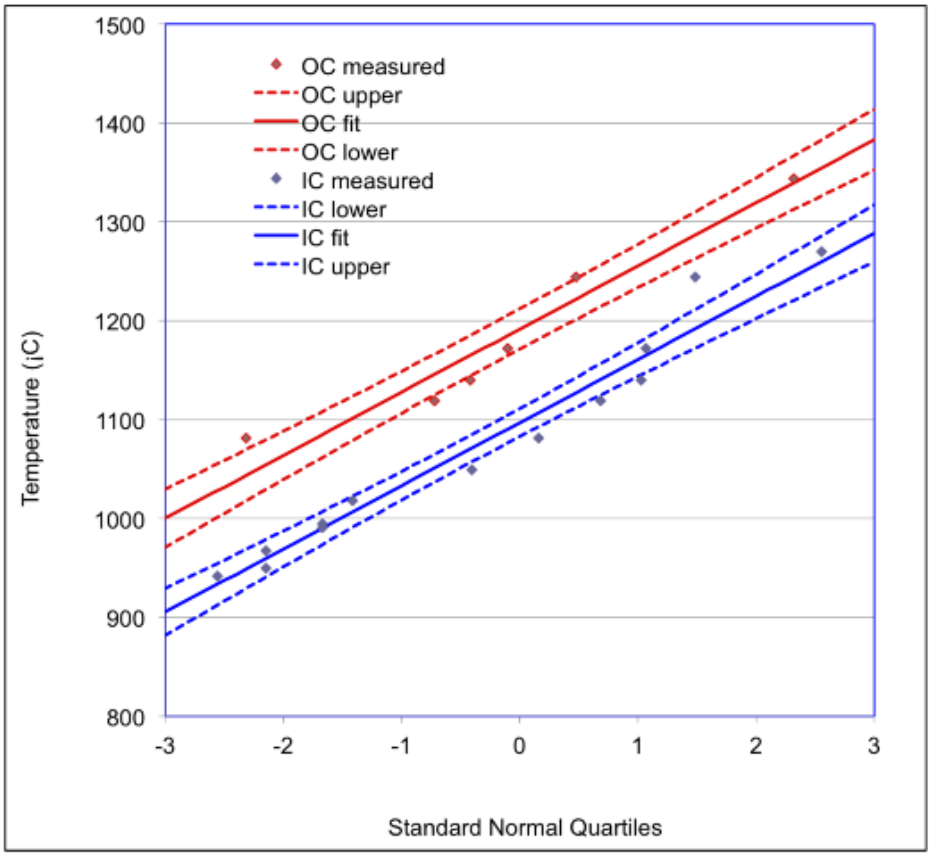

Fig 3.6 Plot of Simultaneous Fit of Outer Core and Inner Core spheres

Plot of Simultaneous Fit of Outer Core and Inner Core spheres yields the two following distributions:

T(OC)= 1191.52 ± 63.84°C

T(IC) =1096.88 ± 63.84°C

[21] H. Nabielek and K. Verfondern, Germany, M. J. Kania, USA Temperatures in the Pebble-Bed: revisiting the 1986 AVR Meltwire Experiment 10 June 2013 IAEA Vienna, Austria

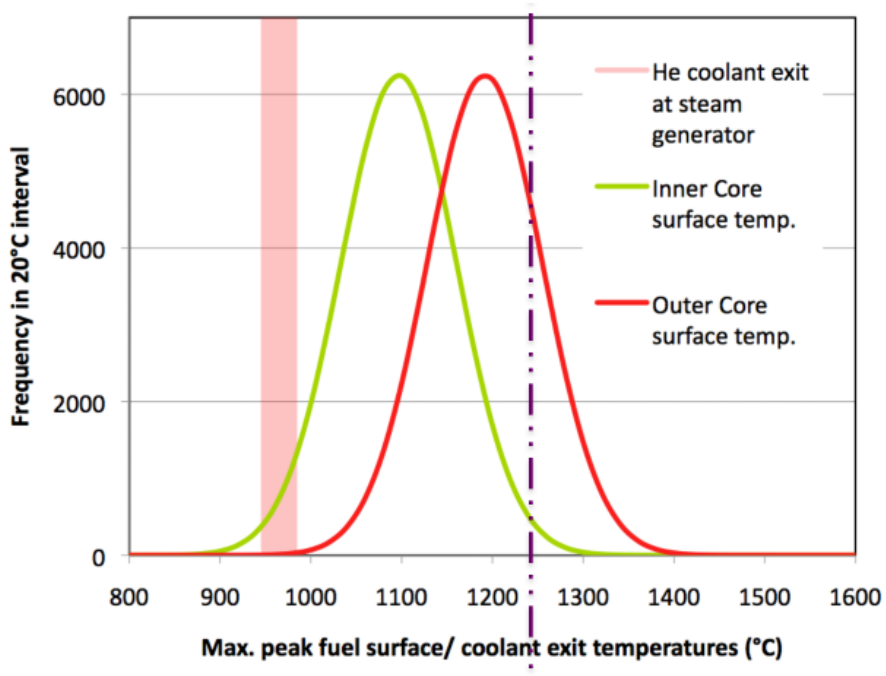

Fig. 3.7 Frequency of core surface and coolant temperatures

Two normal distributions - 1097□64°C and - 1192□64°C define the variation of maximum fuel element surface temperatures in AVR between inner core (green) and outer core (red).

Fuelled sphere center temperatures are higher by a few dozen degrees dependent on operational history, fissile loading and position in the core.[22]

3.2.5 AVR Experience with TRISO fuel

The structure of TRISO fuel element is described in a paper dealing with AVR experience[23]

[22] H. Nabielek… ibid
[23] M. J. Kania*, H. Nabielek**, K. Verfondern; German TRISO Fuel Performance Envelope and Limits – Normal Operations and Accident Conditions, 10 June 2013 IAEA Vienna, Austria

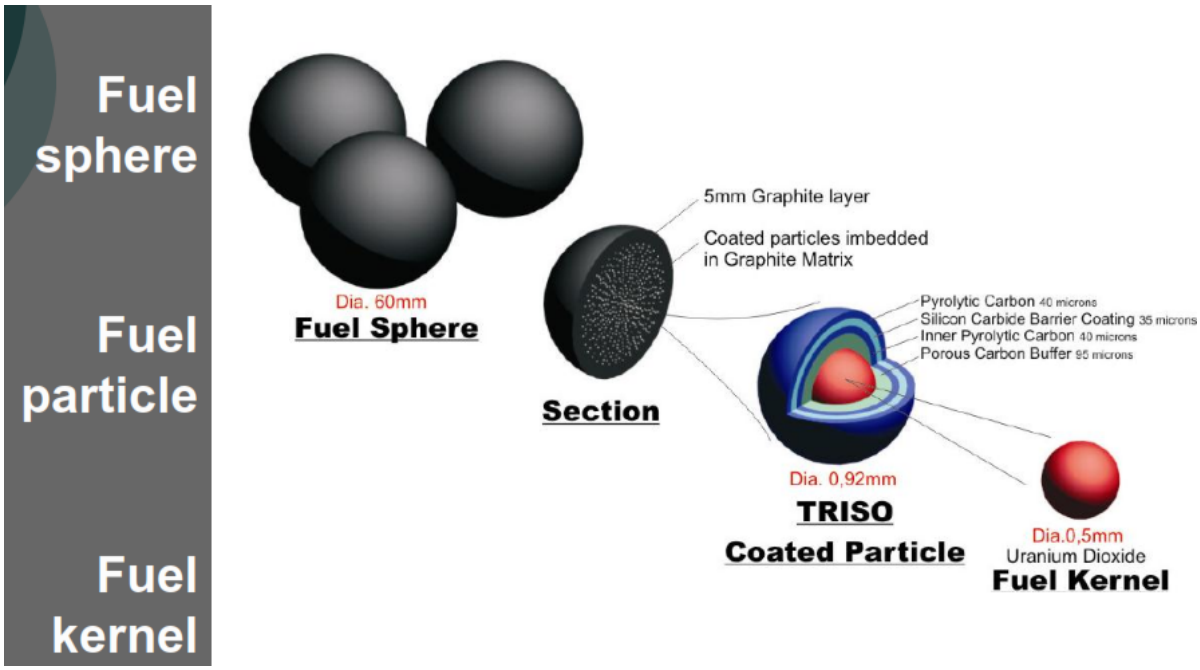

Fig.3.8 Structure of TRISO fuel spheres

Table 3.1 German UO2 and (Th,U)O2 TRISO Particle Designs

TRISO Particle Component	Nominal Dimensions (μm)	
	LEU UO_2	HEU $(Th,U)O_2$
Fuel Kernel Diameter	500	500
^{235}U w/o	9.82 – 16.76	89.01 – 92.46
Th/U Ratio	n/a	4.97 – 10.02
Buffer Layer thickness	95	90
IPyC layer thickness	40	40
SiC layer thickness	35	35
OPyC layer thickness	40	40

IPyC-Inner Pyrocarbon Layer, OPyC -Outer Pyrocarbon Layer

TRISO coating design combines inner and outer LTI PyC layers with a strong, stable SiC layer. This design not only improved clean manufacturing conditions, but also increased the mechanical strength and fission product retention over prior HTI-BISO coated particles.

Table 3.2 Parameters of HTR pebble bed concept

	HTR Pebble Bed Concept		
	PNP-3000	**HHT-Demo**	**HTR Modul**
Operating Parameter			
Duration [d]	1160	1100	1206
Burnup [% FIMA]	9.9 – 11.2	10.4 – 11.2	8.5
Fluence [E>16 fJ, m^{-2}]	$3.1 - 4.5\times10^{25}$	$3.1 - 4.5\times10^{25}$	2.3×10^{25}
Central max. temp [ºC]	1020	1014	870 (Cycled)
Max. FE power [kW]	2.9	2.7	-
Performance Requirements			
Matrix Contamination	3×10^{-5}	3×10^{-5}	5×10^{-5}
As-fabricated defects	6×10^{-5}	6×10^{-5}	1×10^{-4}
Irradiation-induced defects	2×10^{-4}	2×10^{-4}	2×10^{-4}
Accident Conditions	-	-	6×10^{-4}

The maximum central temperature is set as 1020oC, so safely below the temperatures which in AVR resulted in increased releases of fission products.

Irradiation tests covered the series of experiments with LEU and HEU fuel[24], as shown below:

Table 3.3 Irradiation test of LEU and HEU fuel

R&D program	**HEU Program for Process Heat and Gas Turbine Applications**			**LEU Program**
	1977-1981			**1982-2010**
Coated particle	Variant 1	**Variant 2**	Variant 3	**UO_2 TRISO**
	$(Th,U)O_2$ BISO	**$(Th,U)O_2$ TRISO**	UC_2 (UCO) TRISO+ ThO_2 TRISO	
Test goal				
Particle performance	BR2-P24	**BR2-P25**	BR2-P23	**HFR-P4 SL-P1**
Fission product transport in intact particles	FRJ2-P22	**FRJ2-P23**	FRJ2-P24	**FRJ2-P27**
Release from kernel	FRJ2-P25	**FRJ2-P25**	FRJ2-P25	**FRJ2-P28**
Chemical effects	-	-	HFR-P3	**HFR-P5**

[24] M. J. Kania*, H. Nabielek**, K. Verfondern; German TRISO Fuel Performance Envelope and Limits – Normal Operations and Accident Conditions, 10 June 2013 IAEA Vienna, Austria

Fuel Test Elements				
Fuel element performance	HFR-K1	**R2-K12 R2-K13**	R2-K12	**HFR-K3**
FE fission product transport	-	**FRJ2-K11**	FRJ2-K10	**FRJ2-K13 FRJ2-K15**
Large-scale demonstration	AVR 14 AVR 18	**AVR 15 AVR 20**	AVR 13	**AVR 19 AVR 21**
Proof tests				**HFR-K5 HFR-K6**
High burnup tests				**FRJ2-K15 HFR-EU1**
High temperature tests for hydrogen production etc.				**HFR-EU1bis**

Table 3.4 Performance of HEU (Th,U)O2 TRISO based on MTR irradiation tests EOL ^{85m}Kr R/B values[25]

Irradiation test/ capsule or sphere	Beginning of life		Beginning of life		
	$\left(R/B\right)_{BOL}^{Kr-85m}$	Equiv. failed particles	$\left(R/B\right)_{EOL}^{Kr-85m}$	Equiv. in-reactor failed particles	Estimated failure fraction (upper 95% confidence limit)
BR2-P25/ 1-12	3.0×10^{-7} *	3	1.0×10^{-6}	5	5.9×10^{-4}
R2-K12/ 1	3.9×10^{-9}	0	3.2×10^{-8}	0	1.4×10^{-4}
2	3.5×10^{-9}	0	3.4×10^{-8}	0	
FRJ2-P23/ 1	$< 10^{-7}$ *	0	1.4×10^{-7}	0	1.2×10^{-4}
2	$< 10^{-7}$ *	0	1.9×10^{-7}	0	
3	$< 10^{-7}$ *	0	2.3×10^{-7}	0	
4	$< 10^{-7}$ *	0	2.1×10^{-7}	0	
FRJ2-K11/ 3,4	1.7×10^{-9}	0	2.7×10^{-7}	1	2.3×10^{-4}

[25] M. J. Kania*, H. Nabielek**, K. Verfondern; German TRISO Fuel Performance Envelope and Limits – Normal Operations and Accident Conditions, 10 June 2013 IAEA Vienna, Austria

FRJ2-P25/ 2	9.2×10^{-7}	51	1.5×10^{-5}	–	–
R2-K13/ 1 4	2.2×10^{-9} 1.5×10^{-9}	0 0	2.1×10^{-7} 1.9×10^{-7}	0 2	1.6×10^{-4}
Total for entire HEU $(Th,U)O_2$ population (126,664)				8	1.1×10^{-4}
Total for HEU $(Th,U)O_2$ population in 60 mm diameter fuel elements (82,720)				3	9.4×10^{-4}
Total for HEU $(Th,U)O_2$ population in non-standard fuel specimens (43,950)				5	2.4×10^{-4}

*3.2.6 **Observations: In-Reactor Performance***

In-reactor failure levels:

- Results based on accelerated MTR and real-time AVR irradiation tests

- LEU UO2: ≤ $2.1x10^{-5}$

(upper 95% C.L .- 9 failures out of 750,852 TRISO particles examined)

- HEU (Th,U)O2: ≤$6.5x10^{-5}$

(upper 95% C.L. - 8 failures out of 220,990 TRISO particles examined)

- Compared to Performance Requirements:

- ~10X lower HTR Modul specification

- 3X lower PNP/HHT specification

- Majority of particle failures observed in non-reference fuel specimens, at maximum or higher burnups and above maximum fluence requirements

The temperature curve for simulated Dry Loss of Flow Accident[26] is shown below:

[26] M. J. Kania*, H. Nabielek**, K. Verfondern; German TRISO Fuel Performance Envelope and Limits – Normal Operations and Accident Conditions, 10 June 2013 IAEA Vienna, Austria

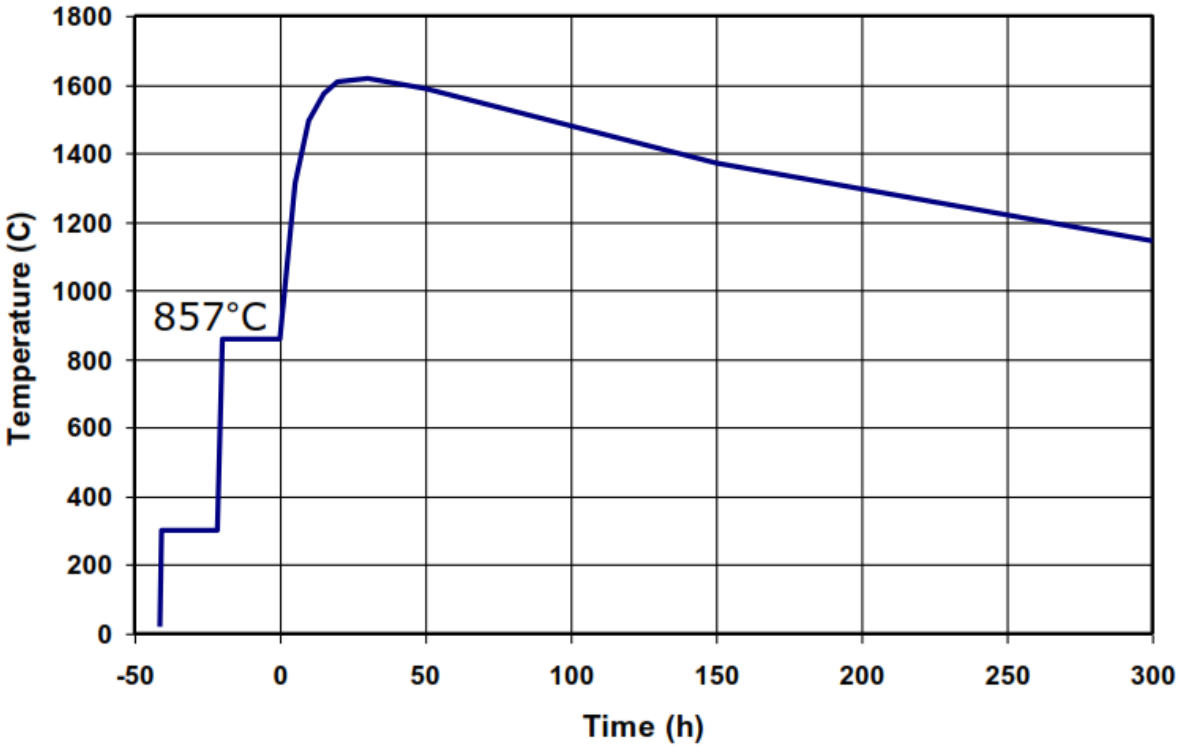

Fig. 3.9 HTR Modul D-LOFC Heating Curve

The temperature under accident conditions reaches 1620 °C.

Fission product releases under accident conditions were determined for two types of TRISO fuel, nmamely AVR fuel and MTR fuel, with the results shown in Fig. 3.10 and Fig. 3.11

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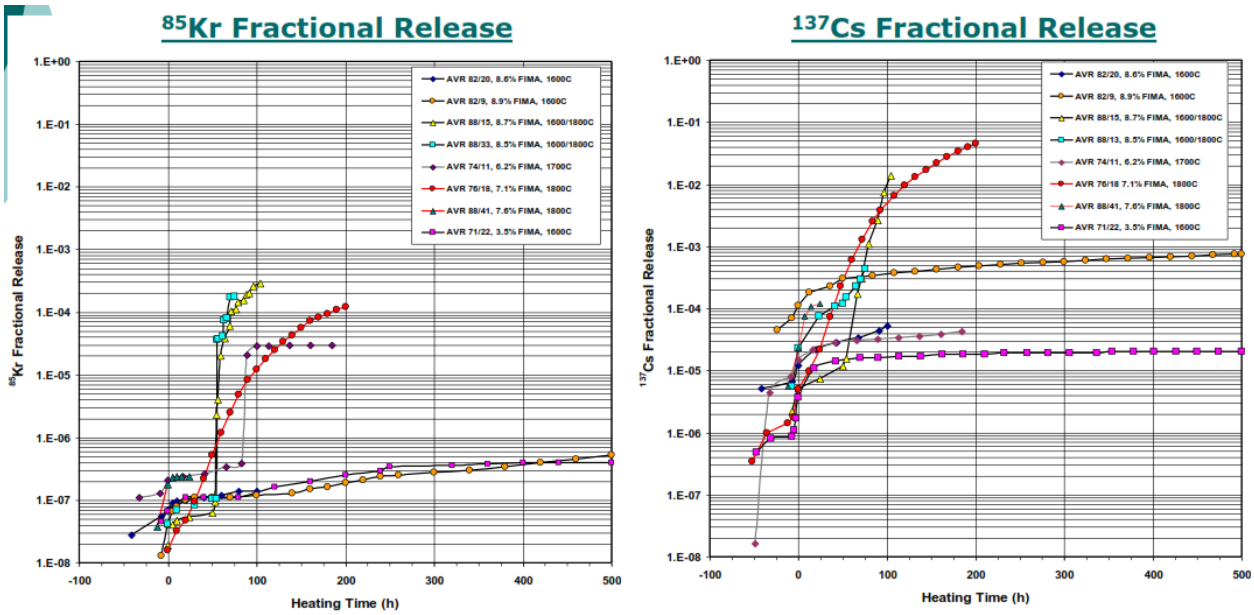

Fig. 3.10 Isothermal Heating Tests on LEU UO_2 TRISO AVR Elements (GLE 3)

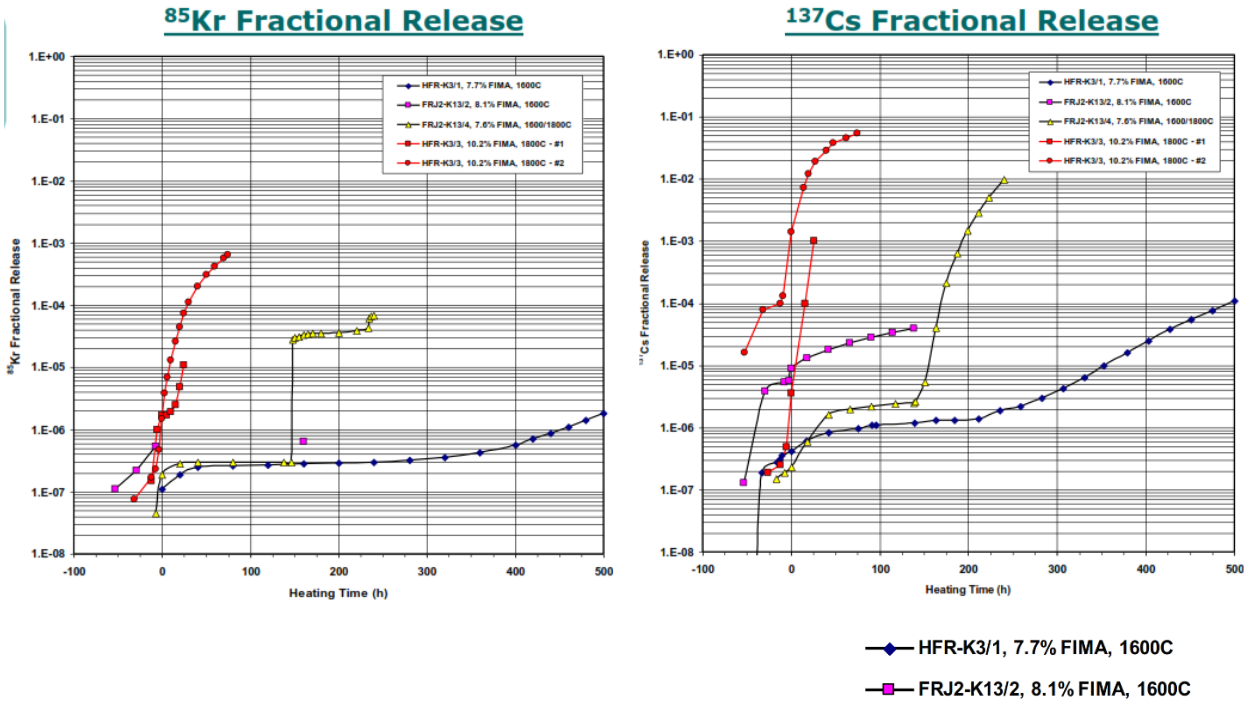

Fig.3.11 Isothermal Heating Tests on LEU UO_2 TRISO MTR Elements[27]

In tests of AVR fuel accidents performed at isothermal conditions at 1600 °C the maximum fractional releases of Cs-137 were 3.6 10-4 after 100 hours and 8 10-4 after 500 hours, while in tests of MTR fuel they reached 3 10-5 after 100 hours and 1 10-4 after 500 hours.

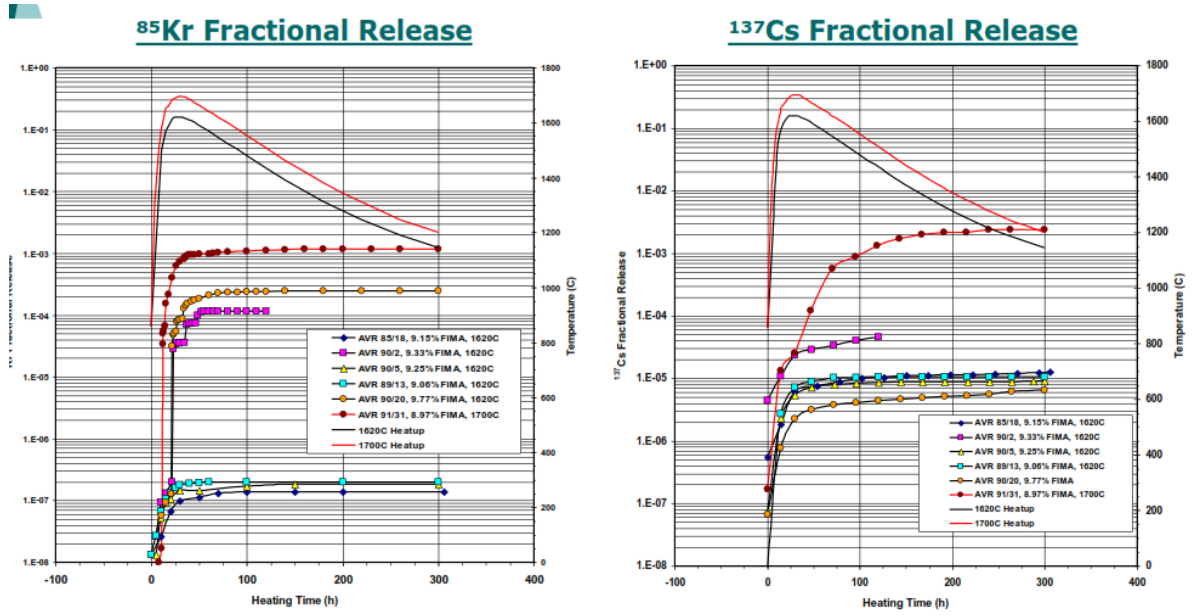

Fig. 3.12 Temperature and release curves in function of time are shown for Kr-85m and for Cs-137

The results of tests GL-4[28] are shown below.

[27] M. J. Kania*, H. Nabielek**, K. Verfondern; German TRISO Fuel Performance Envelope and Limits – Normal Operations and Accident Conditions, 10 June 2013 IAEA Vienna, Austria

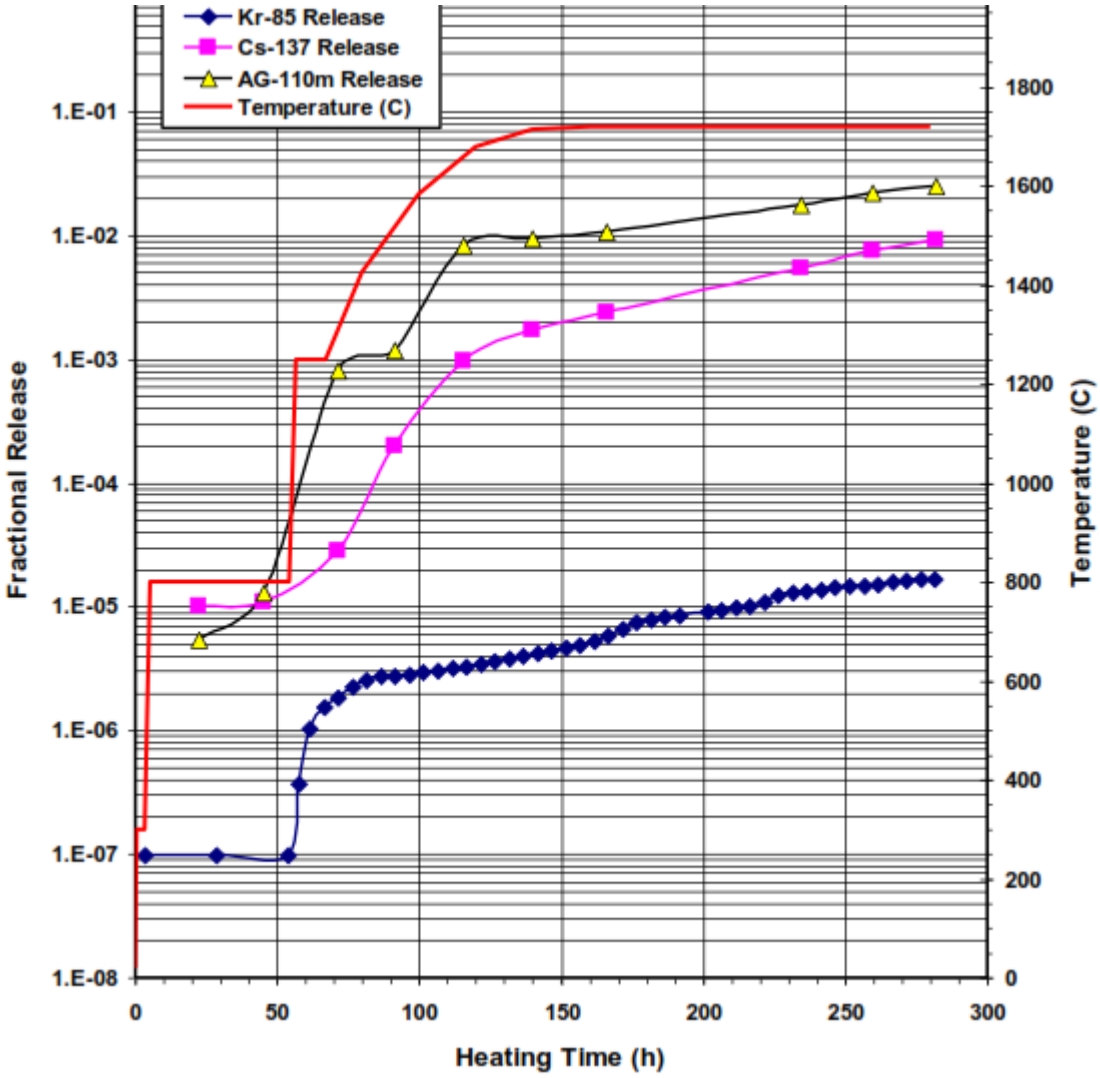

Fig. 3.13 AVR GLE 4/1 Element Tests conducted at ITU, Karlsruhe, Germany

The ratio of release rates of metal and gaseous fission products in these tests is quite different from that typical for light water reactors.

The release rates for Cs-137 and Kr-85m for two other types of TRISO fuel are shown below[29].

In these test the temperature of the core was increased step by step, with temperature reduction to the ambient temperature after each step.

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[28] M. J. Kania*, H. Nabielek**, K. Verfondern; German TRISO Fuel Performance Envelope and Limits – Normal Operations and Accident Conditions, 10 June 2013 IAEA Vienna, Austria

[29] M. J. Kania*, ibid

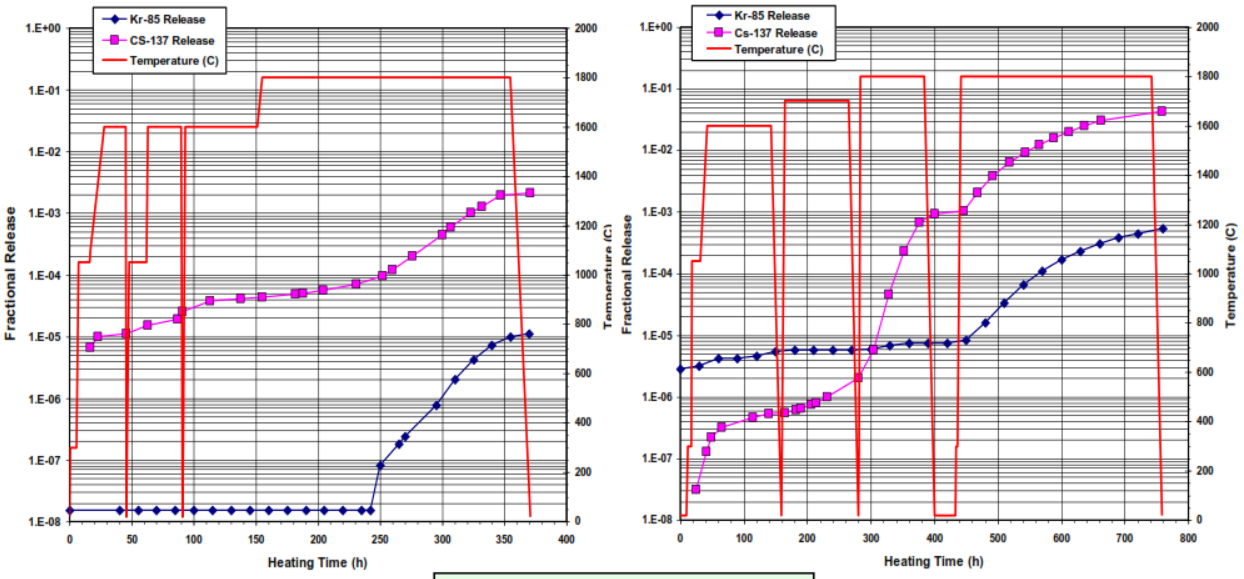

Fig. 3.14 Release rates for Cs-137 and Kr-85m for two other types of TRISO fuel

HFR-EU1bis Isothermal Heating Tests[30]

[30] M. J. Kania*, H. Nabielek**, K. Verfondern; German TRISO Fuel Performance Envelope and Limits – Normal Operations and Accident Conditions, 10 June 2013 IAEA Vienna, Austria

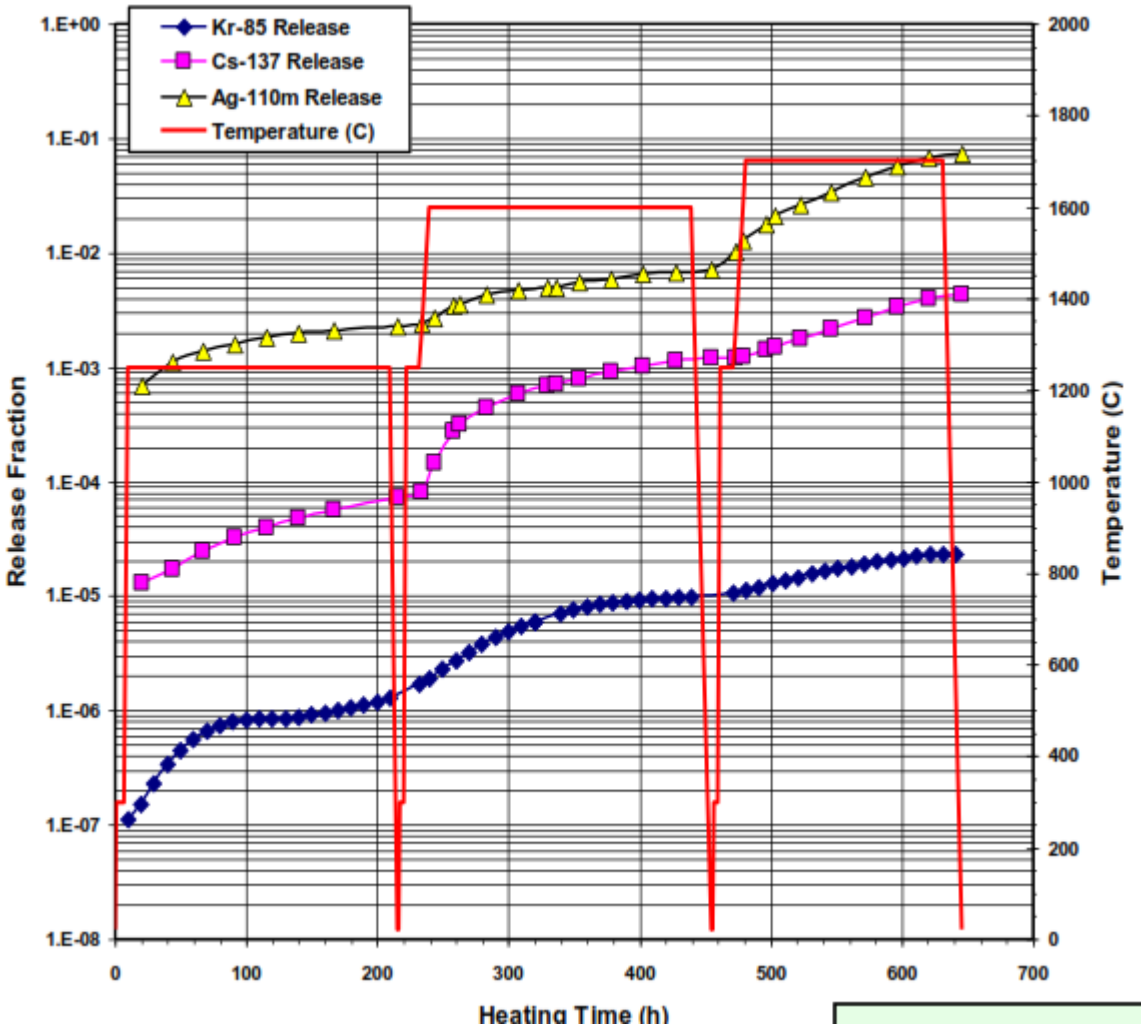

Fig. 3.15 HFR-EU1bis/1, LEU UO2 GLE 4/1 (ITU Accident Simulation Testing), showing releases of Kr-85 Cs-137 , Ag-110m in increased Temperatures (C

The release fraction of Ag 110m after 650 h rose up to 8x10-2. However in the next type of fuel the releases of all fission products were less.

The results for HFR-EU1bis/3, LEU UO2 (ITU Accident Simulation Testing) GLE 4/1 are presented below.

This time the release of AG110m was 3.6 10-3 after 650 hours, and the remaining releases were also less than before, namely for Cs-137 2.6 10-3 and for Kr-85 only 2.2 10-6.

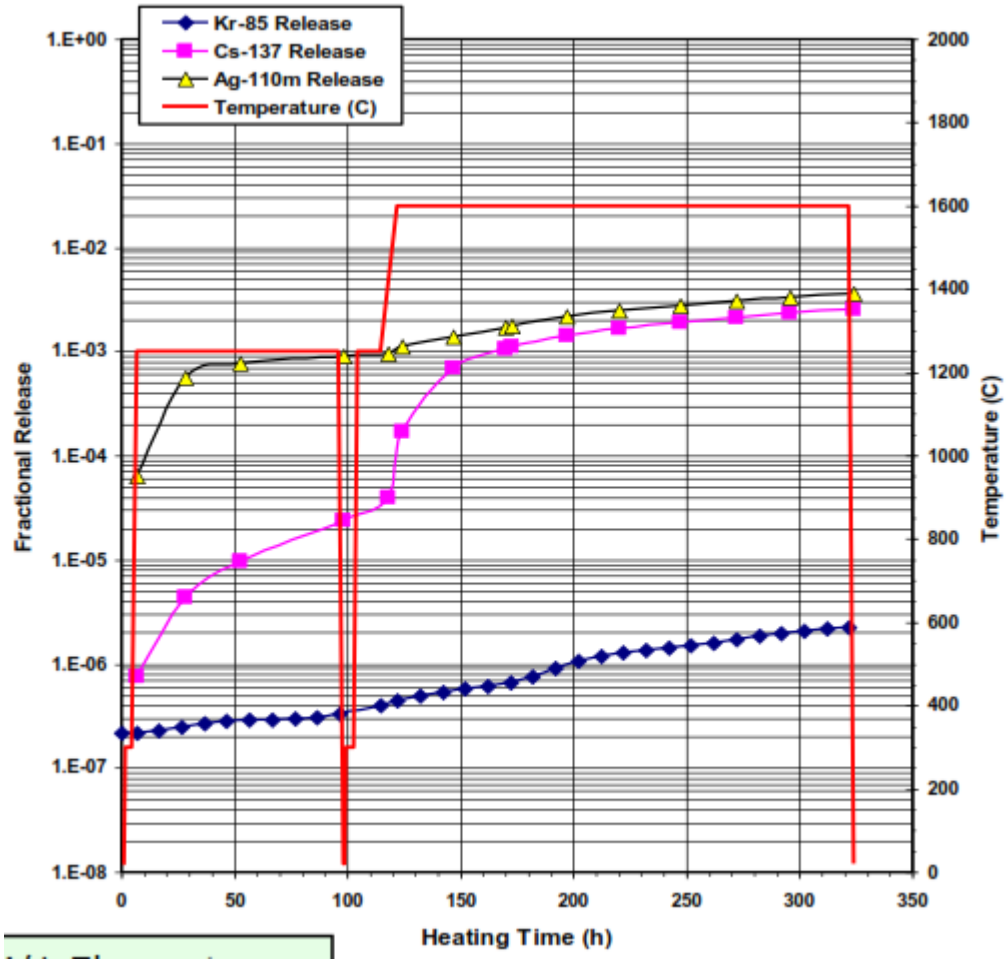

Fig. 3.16 Temperature curve assumed for accident tests of HFR-EU1bis/3, LEU UO2 (ITU Accident Simulation Testing) GLE 4/1 and the resulting fission product release fractions.[31]

Accident Condition Testing Results:

- Full (100%) retention of gaseous/solid fission products demonstrated up to 1600°C for 100 to 200 hours (Note: exception Ag).
- "A-Test" test data are unreliable - because of repeated auto-temperature control failures
- Irradiated MTR elements experience higher temperatures and accumulate higher fast fluences. Some of these elements exhibited active diffusive release of solid fission products (Cs) from intact TRISO particles.
- AVR elements exhibit Cs surface contamination due to presence in the AVR of high-releasing elements from prior poor-quality fuel campaigns.
- Kr fractional release typically lags Cs release because of holdup provided by intact PyC layers in TRISO fuels.

[31] M. J. Kania*, H. Nabielek**, K. Verfondern; German TRISO Fuel Performance Envelope and Limits – Normal Operations and Accident Conditions, 10 June 2013 IAEA Vienna, Austria

3.2.7 *Conclusions*

- Performance, meeting or exceeding HTR Modul requirements, for the modern LEU UO2 TRISO fuel system was successfully demonstrated in 3 primary areas:

• Manufacturing

• In-reactor performance under normal operating conditions

• Accident condition performance

- Performance statistics for the HEU (Th,U)O2 TRISO fuel system are in excellent agreement with those obtained for the LEU UO2 TRISO fuel system.
- Thorium fuel system has the potential for higher burnup and higher accident temperatures, but may require additional testing to achieve (i.e. Accident condition testing).

According to Petti[32], releases of Cs-137 are as follows:

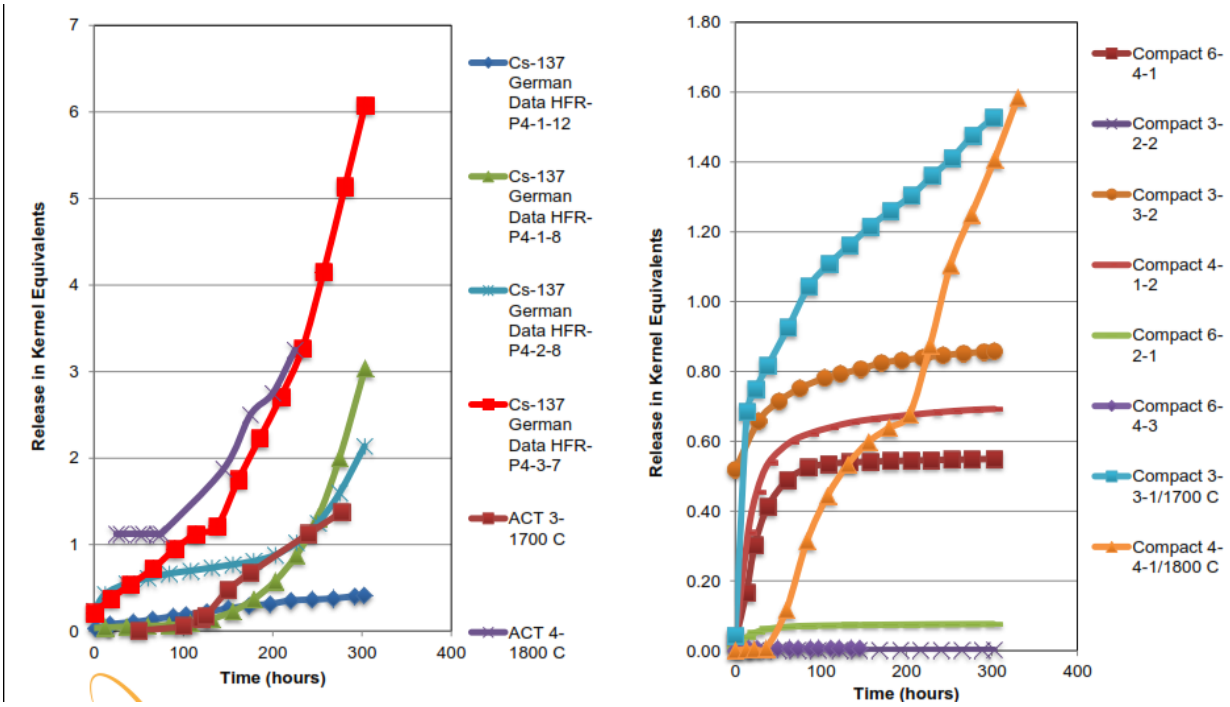

Fig. 3.16 Releases of Cs-137 according to results of Petti

3.2.7.1 *Comparison of Cs data*

• Release from one particle looks diffusional but it is very different than when more than one particle has failed SiC or is a failed particle

• Consistent release per failed SiC between US testing and HFR-P4-1-12 at 1600°C.

– HFR-P4-1-12 suggests 0.5 kernel equivalents at 1600 C; AGR 6-4-1 and 4-1-2 suggests 0.55 to 0.69

• Greater Cs release in other German and Japanese heating tests.

[32] Petti, ibid

– Shape of the UO2 TRISO release curves are different than UCO TRISO

– Releases of 2 to 6 kernel equivalents in German tests suggest SiC failures of 4 to 12

• US and Japanese releases 1700 C data look similar

– 1.5 kernel equivalents

– Given 2 particles had failed SiC in US tests, this implies ~ 2 SiC failures in ACT particles and 0.75 Cs release at 1700°C. IMGA data from ACT particles suggest 3 to 4 particles released some Cs

• Similarity between ACT-4 and HFR-P4-3-7 suggests probably more than one completely failed particle

3.2.7.2 Comparison of Sr release data

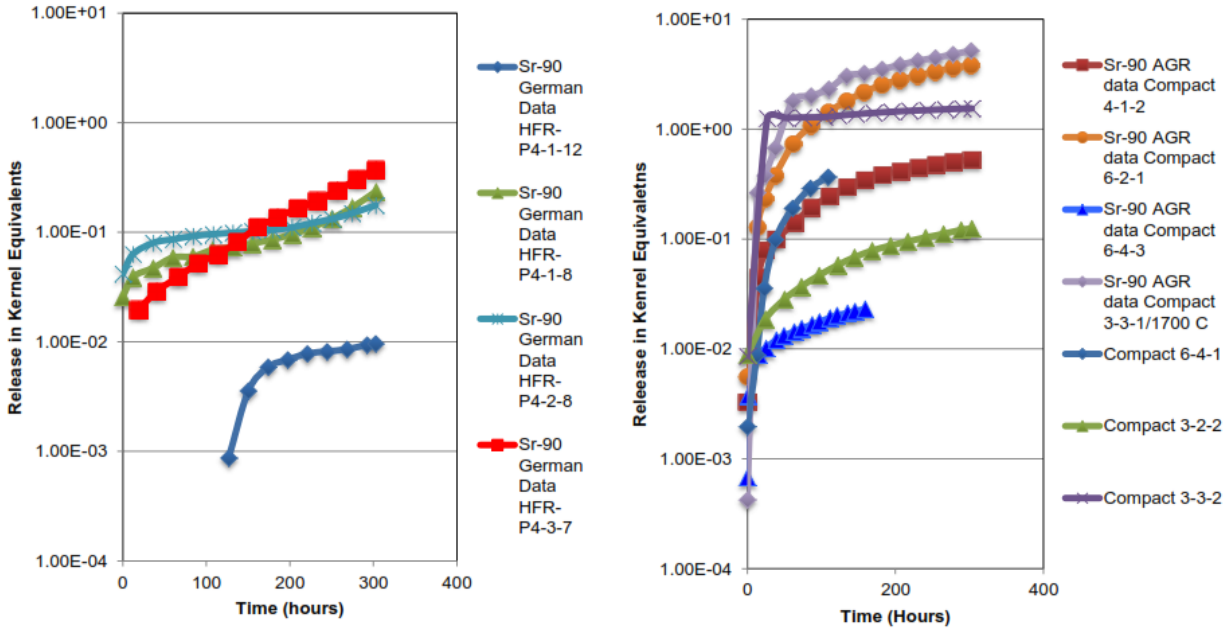

Fig. 3.17 Comparison of Sr release data

3.2.7.3 Comparison of Eu data

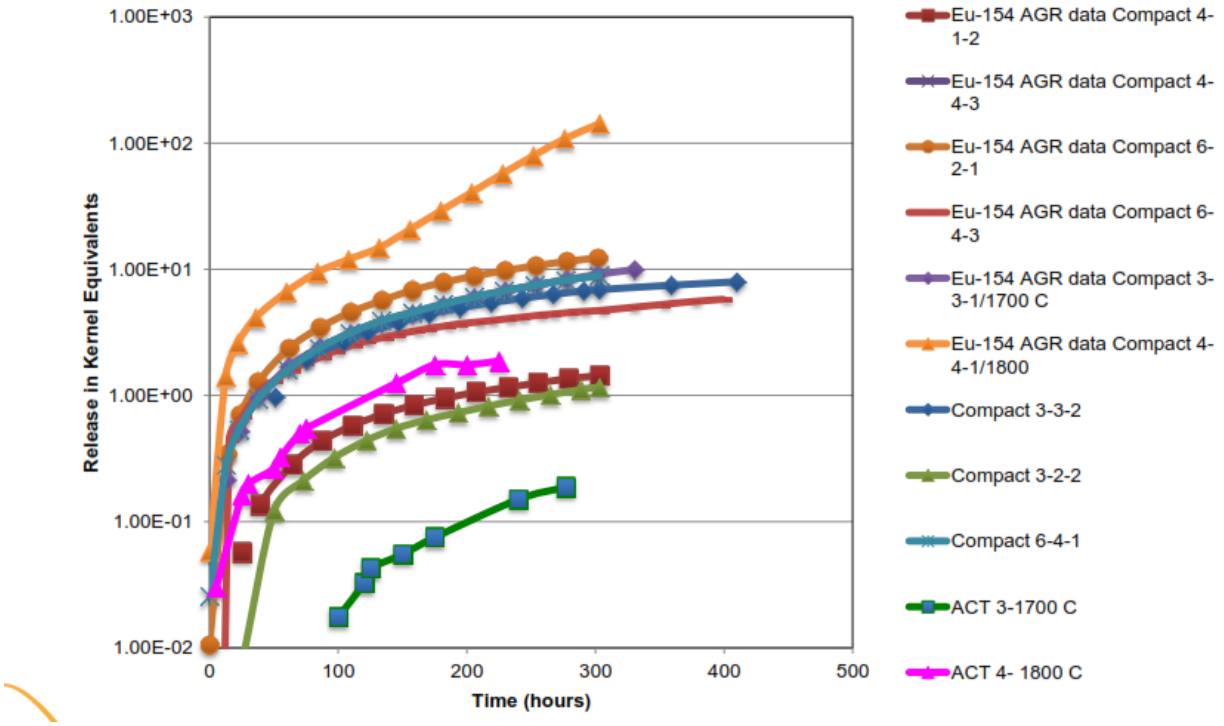

Fig. 3.18 Comparison of Eu data

3.2.7.4 Comparison of Kr data

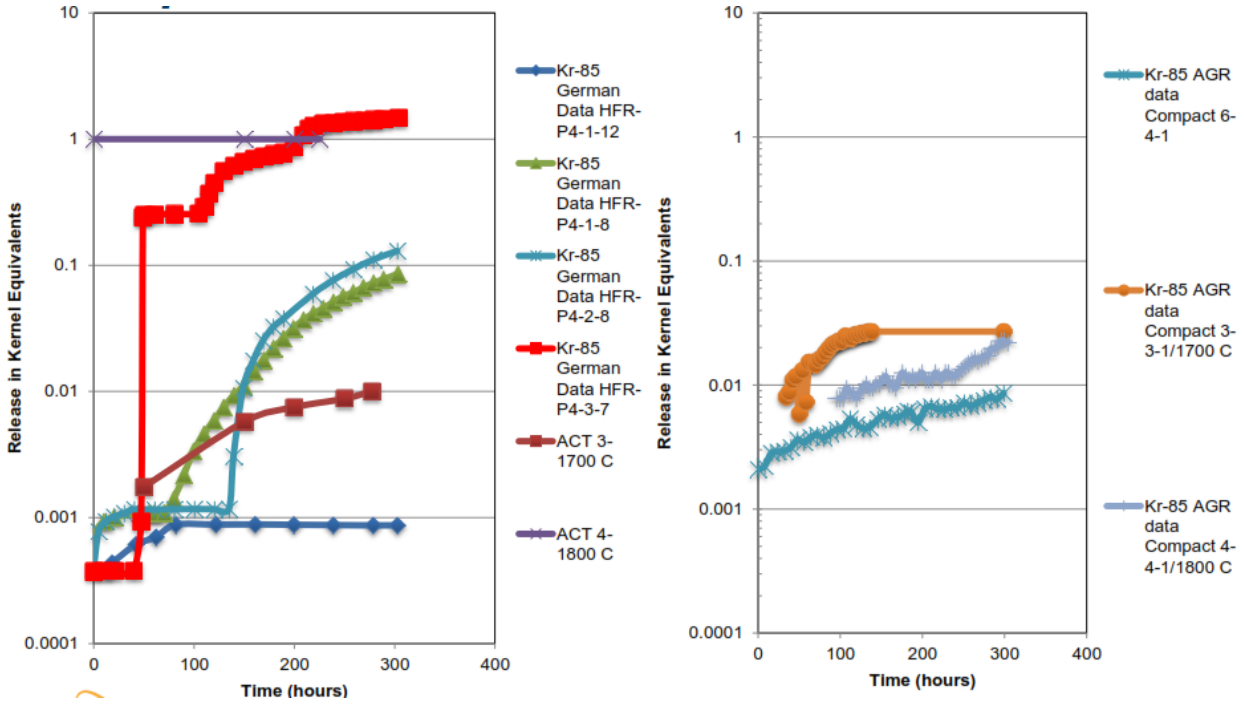

Fig. 3.19 Comparison of Kr release data

Release depends on how many particles had SiC failures.

• Some similarity in gas release when different kernel sizes are taken into account

– Results from US AGR Compact 6-4-1 are three times HFR-P4-1-12 at 1600°C

– Results from US AGR Compact 3-3-1 (with two SiC failures) are similar to ACT3 at 1700°C

• Appears that German tests had many SiC failures (based on Cs results) and some particle failures

3.2.7.5 Failure mechanisms

• US UCO TRISO failures are due to cracking of buffer leading to crack in IPyC and SiC

• Japanese UO2 TRISO particles indicate buffer cracks and IPyC and SiC cracks

• Germans attribute high releases to increased SiC permeability due to fission product attack

Complex behavior observed because of superposition of release from intact particles, release from particles with failed SiC and releases from failed particles

• Release behavior (timing and magnitude) and physical appearance of UCO TRISO appears to be different from UO2 TRISO, especially at 1800°C

- Degradation of SiC layer in UO2
- Releases of Kr and Cs from UO2 TRISO but not in UCO TRISO are greater than from UCO TRISO
- Releases of Sr and Eu from UCO TRISO are greater than from UO2 TRISO

Influence of chemistry of the kernel:

• CO attack of SiC

• Oxygen potential in fuel is very different in UO and UCO TRISO

• PARFUME calculations are planned to help sort out behaviour

The presentation of Petti et al.[33] shows conclusions reached on the basis of AGR-1 tests

3.2.8 Summary of conclusions from UCO TRISO Fuel Safety Testing

• Accident safety testing of UCO TRISO from AGR-1 is nearing completion and demonstrating robustness of fuel

– Releases are very low unless a defective particle is present or a particle fails during testing

– Releases are from fission products that diffused into the matrix during the irradiation and not from the intact TRISO particles during the high temperature heating

– No through particle failures at any temperature

– UCO TRISO fuel is able meet in-service failure fraction under off-normal conditions

[33] D. Petti et al.: What does AGR-1 tell us About Temperature Limits of UCO TRISO Under Accidents?

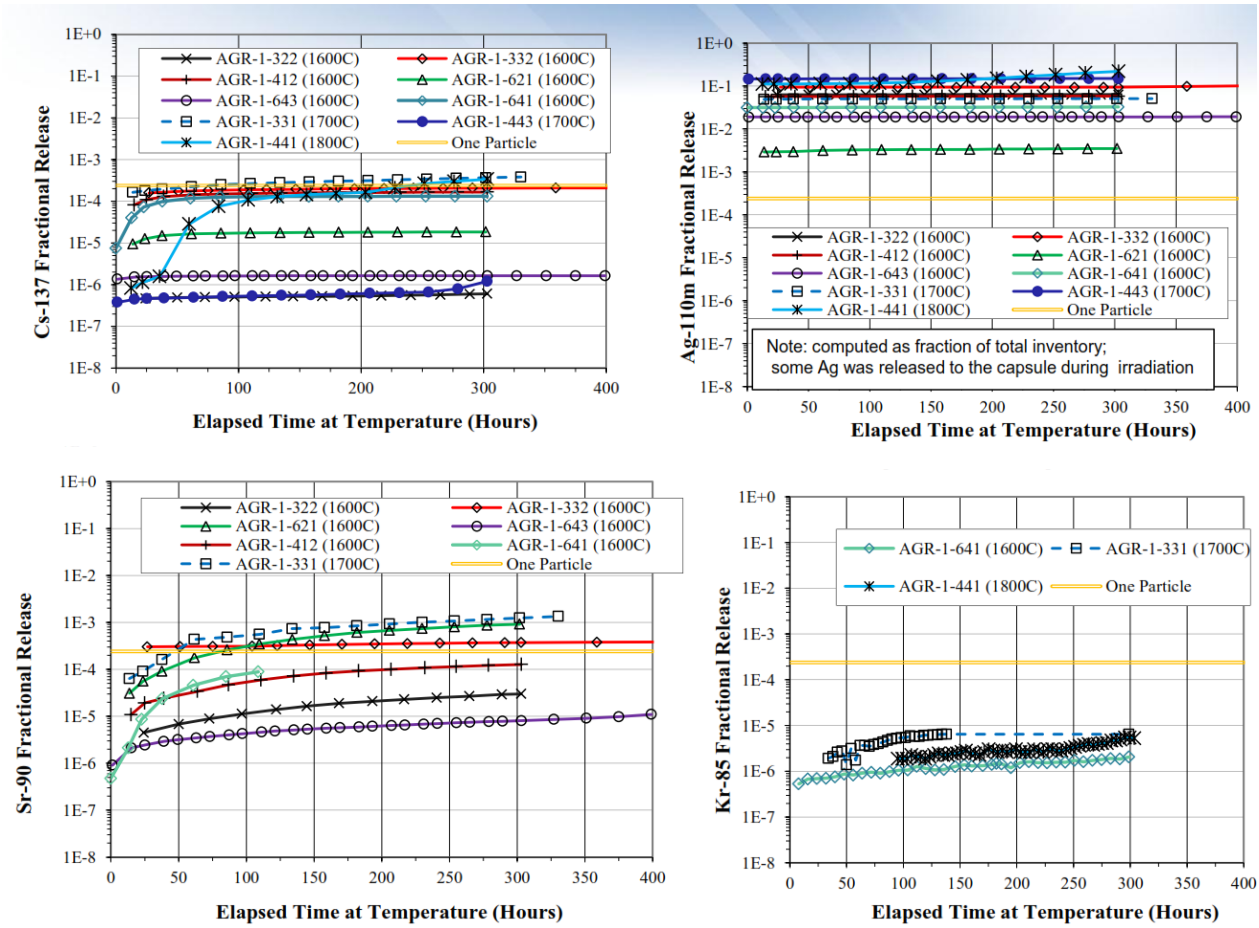

Fig. 3.20 Fractional releases, measured in isothermal conditions in AGR experiments

After keeping the test fuel at isothermal conditions at 1600 oC, the maximum release fraction of Cs-137 was 3 10-4 after 300 h, Ag-110m 0,1 very soon, already after 20 h, Sr-90 after 300 h was 10-3, and that of Kr-85 was 2 10-6 after 300 hours.

Accident safety testing of UCO TRISO from AGR-1 is nearing completion and demonstrating robustness of fuel

– Releases are very low unless a defective particle is present or a particle fails during testing

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– No through particle failures at any temperature

– UCO TRISO fuel is able to meet in-service failure fraction under off-normal conditions

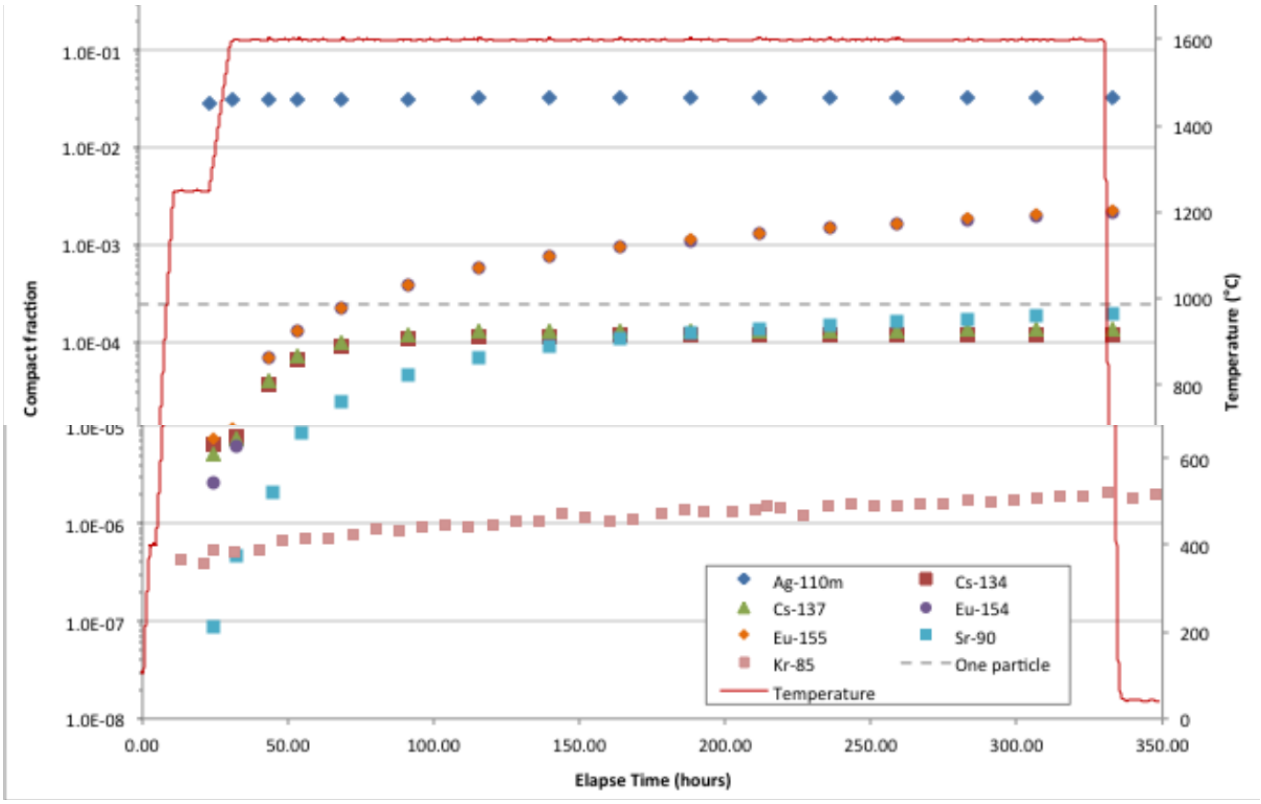


Fig. 3.21 Results of Compact tests

Preliminary results of test COMPACT 6-4-1 13.2% FIMA 1041°C TAVA Temp

• Preliminary results from fission gas system and all condensation plates

• Rapid release of ~3% of Ag-110m at 1250°C but no release after ~22 h at 1600°C

• Release of Cs once temperature reaches 1600°C but rate decreases significantly by ~100 hours. Fraction released = 1.2E-4. Indicates possibly one particle with failed SiC

• Kr-85 release rate is low and constant throughout test. No failed TRISO

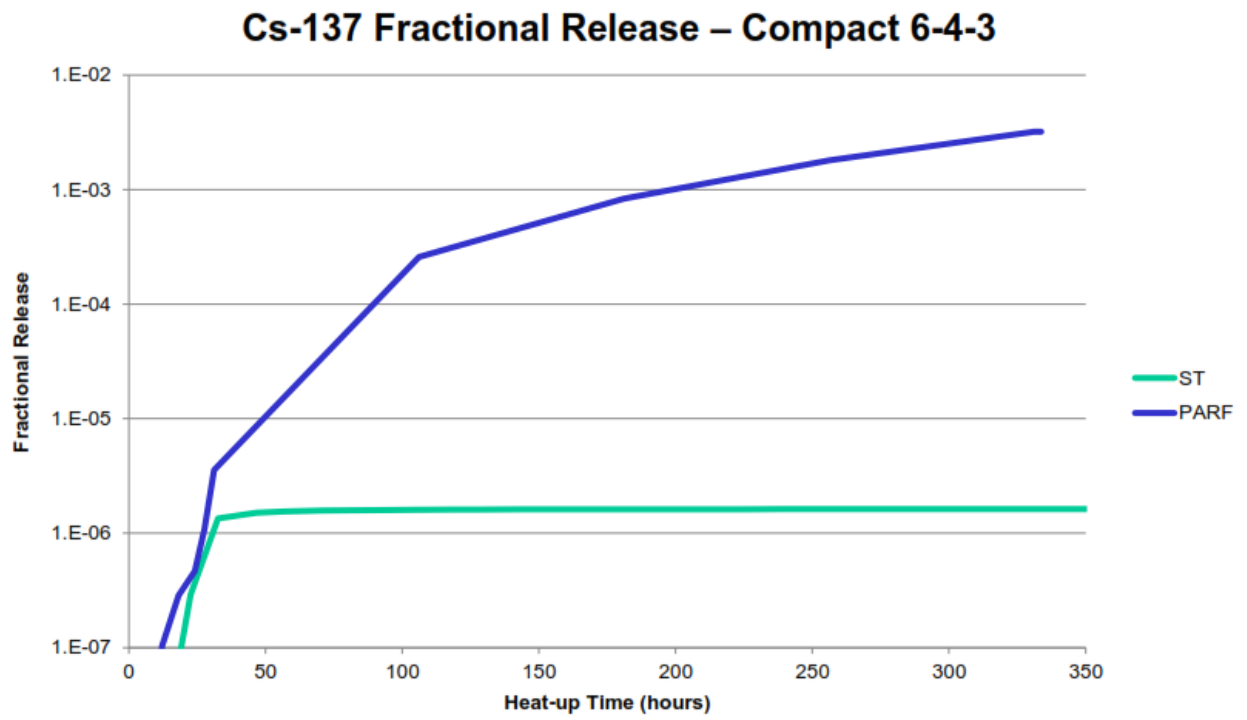


Fig. 3.22 Fractional release of Cs 137

3.2.9 Cs release from AVR

AVR end-of-life Cs-137 core release has been experimentally determined at (0.2+-0.1)% of all Cs-137 generated through the lifetime of the reactor. A conservative estimate based on 100% caesium release from all contamination and all particle defects and failures yields the AVR end-of-life Cs-137 core release prediction at 1.49%. Assuming maximum fuel temperature peaks of 1450°C, the additional release through intact TRISO particles increases the predicted core release fraction to 1.51% is therefore negligible.

3.2.10 Conclusions on HTGR fuel behavior

Information gathered from extensive irradiation and heating tests has provided a fundamental understanding of the performance capabilities of present day TRISO fuel and has allowed advances in the modeling of fuel performance. Several performance limiting mechanisms for particle failure (i.e. the significant release of fission products from TRISO particles) have been identified; these as well as means for controlling them are given below.

Pressure induced failure. The buildup of pressure inside the particle coatings due to the generation of fission gases results in a tensile stress in the SiC load bearing layer. If this stress exceeds the strength of the layer, the result is a simultaneous failure of all the coating layers. Pressure induced failure is controlled by product specifications on the distribution of fuel kernel diameters, coating thickness and on the 'void' volume provided in the buffer layer.34

[34] IAEA TECDOC 1198 ibid

Kernel-coating interactions. Carbon transport in the presence of a thermal gradient may result in kernel-SiC layer contact and coating degradation. Design selections of core operating temperatures and associated temperature gradients limit the kernel-coating interactions to acceptable values.

SiC dissociation and increase in SiC porosity at high temperatures. Thermal damage and/or increased porosity of SiC may occur at very high temperatures, resulting in increased fission product release after extended times. Thermal dissociation can lead to complete SiC degradation at 2200°C. To mitigate these effects, reactor design conditions can be selected to keep fuel temperatures below 1600–1700°C under severe accident conditions, at which temperatures the above mechanisms have a negligible effect on fuel coating failures.

(d) SiC-fission product interactions. Fission products that are released from the kernel and reach the SiC layer may interact chemically with the SiC and result in its degradation and, ultimately, in coating failure. Testing under severe conditions with high thermal gradients specifically intended to concentrate fission products at the SiC surface has shown that palladium (a fission product) interacts with and degrades the SiC. To mitigate these effects, reactor design parameter values are selected to limit core thermal gradients and maximum temperature to values such that SiC-fission product interactions are negligible.

(e) Irradiation effects on coating integrity. If binding occurs between the fuel matrix material and the outer coating of the fuel particle, irradiation stresses can lead to coating failures due to the relatively high shrinkage of matrix material and the 'tearing away' of portions of coating layers during irradiation. Such types of coating failures can be avoided by designing the fuel element so that only a weak physical coupling exists between the outer coating of the fuel particle and the surrounding matrix material, under which conditions the above mechanism is insignificant. Also, dimensional changes at the PyC layers under irradiation has led to fracturing of layers, delamination and damage to the SiC layer. This is addressed by controlling the coating conditions.

For normal reactor conditions, irradiation testing of high quality fuel has been
performed in material testing (or research) reactors, as well as operating HTGRs. Parameters such as heavy metal burn-up, operating temperature, and fast neutron fluence are varied to assess fuel performance. Continuous monitoring of released fission gases during irradiation tests gives a direct indication of the integrity of fuel coatings. The deconsolidation of fuel elements and subsequent individual particle gamma spectroscopy allow the determination of the fission product retention (fuel performance) variabilities within a large population of particles.

In the early 1980s, both the USA and German fuel programmes embarked on improving development based on the TRISO coating design. The German programme focused on the UO2 kernel, whereas the US programme concentrated on the UCO kernel.

Investigation into the UCO kernel by the USA was initiated in an attempt to achieve high burnup of low enriched fuel. Revisions in the US regulatory requirements mandated a change in the enrichment level to a maximum of 20%. Studies indicated that fuel economics improved with an increase in enrichment. However, systematic and technical developmental problems within the US programme prevented the achievement of the UCO kernel to reach the intended enrichment/burnup goals.

Within Germany, a systematic incremental improvement programme on UO2 kernel development continued in the 1980s with the top enrichment level at 10%. Nine irradiation tests comprising over 200 000 coated fuel particles were performed; no coating failures were observed as a result of in-reactor gas release. Statistically, the above corresponds to a median probability (50% confidence level) that the fuel coating failure fraction in a much larger batch of coated particles is less than $3\ 10^{-6}$, and to a 95% confidence level that the coating failure fraction is less than $1\ \_10^{-5}$. Calculated failure fractions based on fuel performance models have been obtained for MHTGR normal operating conditions, and

are in the range of 2 to 5 _10^{-5} . German fuel utilizing the UO2 kernel now forms the basis for fuel development of all HTGR programmes.

Currently, only Japan has the fuel fabrication facility to provide limited quantities of coated particles on a commercial basis. The HTTR fuel from this facility consists of TRISO coated UO2 kernels, 600 μm in diameter. The quality of the first loading of HTTR fuel exhibited an average bare uranium fraction and SiC defective fraction of the fuel compacts at 2 10^{-6} and 8 10^{-5}, respectively.

Fuel testing under off-normal conditions has provided fuel performance information as a function of fuel temperature, up to 2500°C. Radiologically, the most significant nuclide released as a gas from the fuel is iodine. Fractional krypton release is easier to measure than fractional iodine release and gives a conservative measure of fractional iodine release. The German programme measured krypton fission gas releases from high quality fuel during temperature ramp tests. Each test involved heating an AVR fuel pebble to as high as 2500°C, with each pebble containing 10 000 to 17 000 particles. Fuel temperatures of about 2200°C were required before fission gas release fractions approached the equivalent of one particle coating failure.

Not all fission products are gases. From irradiation and annealing experiments, the sequence of fission product release from HTGR fuel as a function of temperature is generally observed to be, in order: silver, caesium, strontium, gases (e.g. krypton), cerium and ruthenium/zirconium. Specific US results from a temperature ramp test based on testing fuel particles by themselves, show that the fission gas release results (i.e. krypton) are in good agreement the results for AVR fuel, which illustrates that the fission gases are being retained within the coated fuel particles to very high temperatures (~2200°C). Silver isotope, Ag110m, is not retained well at high temperatures and diffuses through SiC and pyrocarbon. The amount and level of influence this isotope has on the metallurgical and maintenance considerations for the power conversion system components on the closed cycle gas turbine HTGR plants over the long period of operation is not yet fully determined. Also, if operation is allowed to exceed a fuel temperature of 1250°C. over an extended period of time, SiC coating thickness deterioration will occur due to palladium.

4 Plant response to postulated initiating events (PIE)

4.1 Postulated initiating events for safety analyses of HTGR[35]

A unique aspect of the MHTGR is that design features and parameters have been selected so as to minimize the need for reliance on active safety components such as pumps, motors, valves, and associated support systems. In particular, the core size, geometry, and power density have been selected such that decay heat can be removed from the core solely by the inherent mechanisms of radiation and conduction, thus eliminating any reliance on forced coolant convection, or even the need for coolant, to prevent a significant radionuclide release from occurring. Additionally, the fuel type and enrichment have been selected so as to favor an intrinsically strong negative temperature coefficient, thus the reactor tends to inherently shut itself down in the event of undercooling or overpower transients. This combination of features results in a design which displays an unusually high level of safety.

[35] Probabilistic Risk Assessment For The Standard Modular High Temperature Gas-Cooled Reactor DOE-HTGR-86-011 Revision 3 Volume 1.

The PRA assessment examined a broad event spectrum in order to identify events potentially dominant with respect to plant safety. From this examination, seven initiating events were selected for detailed evaluation:

1. Primary coolant leaks.

2. Loss of main loop cooling.

3. Seismic activity.

4. Loss of offsite power and inadvertent turbine trip.

5. Anticipated transients requiring reactor scram.

6. Control rod group withdrawal.

7. Steam generator leaks.

From these seven initiating events only primary coolant leaks, seismic activity, and steam generator leaks were found to result in potential offsite releases. The fission product release scenarios include depressurization of the reactor vessel under dry and wet core conditions with or without forced cooling. The accidents under dry conditions are initiated by primary coolant leaks and earthquakes. The accidents under wet conditions are initiated by the steam generator leaks. In these accidents the core cooling can be provided either by one of two forced cooling systems or by conduction through the reactor to remove heat out to the reactor cavity cooling system.

A review of the assessment results confirms the selection of the Licensing Basis Events included within the PSID to be appropriate and consistent with this latest study. The PRA results confirm that even when a broad range of accidents that cover both a large cross section of initiating events and an extreme frequency spectrum is considered, the assessment of plant risk shows the MHTGR to be insensitive to failures in active and engineered systems. The frequency of potential radioactivity releases is essentially dictated by the failure of passive structures in the MHTGR. By virtue of its high reliance on passive features and inherent characteristics in this small MHTGR, the overall safety of the concept is shown to display unusually high levels of safety. The concept is shown to comply with the risk limits of the NRC Safety Goals and to do so with substantial margin. The MHTGR is even shown to satisfy the very stringent user-imposed requirement that Protective Actions Guide (PAG) doses related to public evacuation and sheltering are met at the *425* m site Exclusion Area Boundary. PRA results demonstrate that releases with frequencies as low as *5* x 10-7 per year are below the PAG sheltering limits of 10 mSv whole body and 50 mSv thyroid at the site EAB.

It should mentioned, that the PAG sheltering limit of 10 mSv whole body corresponds to the limit dose in Poland at exclusion area boundary.

The scope includes frequency and consequence assessments for a wide spectrum of events with. frequencies greater than one in one hundred million years (10-8 per year). The assessment considers both events in which the system failure probabilities are not strongly coupled (e.g.,loss of main loop cooling) and two of the most important external events (loss of offsite power and earthquakes) in which the initiating event can simultaneously threaten multiple plant.

The analysis begins with the selection of initiating events.

Initiating events are events which disturb the plant from its normal states, either operating or shutdown sufficiently to result in conditions which could culminate in a release of radioactivity. With regard to fuel inventory, initiating events affecting this source of radioactivity would also likely lead to conditions requiring a plant trip (i.e., either a reactor trip, turbine trip, or both). The selection of initiating events considers each radioactivity source in the plant.

Events are postulated which may defeat or degrade the barriers to release from each source. Plant outage causes are also evaluated as candidate initiating events, since they disturb the plant from its normal power producing state.

In this study, accident sequence development was curtailed if it was determined both that the sequence had a frequency of less than 10^{-8} per year and that the projected consequences of the sequence did not greatly exceed (i.e., orders of magnitude larger) other events to be analyzed. This cutoff guideline is believed to be justified for two reasons. First, the selected frequency is sufficiently below the NRC's mortality safety goal target of 5×10^{-7} per year that and cumulative residual risks associated with events below that frequency level would be adequately included in the final assessed risks. Thus the level of risk assessment is deemed suitable for determining whether or not the NRC's safety goal has been met, a key objective of the risk assessment.

Secondly, it is believed that a frequency of 10^{-8} per year represents a level beyond which the risk assessment result lose useful meaning.

The consequence evaluation of each representative event sequence consists of four parts: thermal-hydraulic transient analysis, radionuclide transport analysis, dose and public health impact assessment, and uncertainty analysis. Thermal-hydraulic transient analyses are used to determine key component temperatures, coolant pressures, and flow rates to establish the mechanisms, timing, and the driving forces for radionuclide transport from the initial location across the barriers to release. The radionuclide transport analysis utilizes the thermal-hydraulic analysis results as input to define the driving forces (i.e. temperatures, flows, shear forces) for release. The radionuclide transport analysis then models the time-dependent release through the radionuclide barriers, properly accounting for radioactive decay and removal mechanisms such as gravitational settling and plateout. The dose and public health impact assessment utilizes the estimated time-dependent radionuclide releases as input and determines the expected impact on a hypothetical individual at the Exclusion Area Boundary (EAB) in terms of dose to the whole body and major organs. These doses are then scaled to obtain the estimated latent fatality risk of this hypothetical individual.

4.2 Control of radionuclides

The sources of radiation must be controlled in the plant. Three principal sources are identified: (1) the reactor core itself, (2) radionuclides in process systems such as the steam, feedwater, and condensate system, and (3) radiation stored in gas, liquid or solid waste systems. Table 4-1 compares the activity stored in these various sites. It may be seen that, by far, the largest activity site is the reactor core itself.

Additionally, the design of the MHTGR presents no unusual challenges to controlling the risks from these other sources of activity.

Table 4-1 Magnitude of activity sites in the MHTGR

Activity Site	Approximate Magnitude (in curies)	
Reactor core	2×10^{9}	Design core equilibrium
Primary coolant system	3×10^{3}	Design circulating and plateout
Secondary coolant system	1	Tritium in secondary water
Helium purification system	5×10^{3}	Design purification
Gaseous waste	2×10^{2}	Regeneration of low-temperature absorber
Liquid waste	1×10^{2}	Decon drains from hot service facility
Solid waste	2×10^{6}	Spent fuel

Risk assessment activity has focused on core activity releases and has not further developed the risks from other sources (assessments performed later in the development of the MHTGR design will be required to confirm that the risks from these other sources are acceptably low, however).

The fourth level in the tree identifies that control of radiation from the core involves the control of both direct radiation (i.e., containment of gamma or neutron energy release) and transport of fission products from the core. The offsite public is inherently protected from the former hazard by the sub grounding of the reactor and distance from the source. Any threat which could remove this inherent protection is considered extremely remote. The risk assessment, therefore, has only focused on events which affect the possibility of activity transport from the core (although any future consideration of risks to operations will have to consider direct radiation consequences).

The fifth level identifies four functions associated with the barriers to fission product release from the core: (1) control release from the core itself, (2) control the transport from primary circuit, (3) control transport from the reactor building, and (4) control transport from the site. Conventional reactor designs have placed considerable emphasis on all these functions, but especially the third function. Accordingly, the performance or nonperformance of the reactor building in conventional designs has a significant influence on assessed risks. As has been discussed in earlier sections, however, the safety philosophy of the MHTGR has been to place primary reliance on the achievement of the first function to minimize risks. Intentionally less reliance has been placed on other functions (barriers). Accordingly, the risk envelope of the MHTGR is projected to be less sensitive to failures of barriers beyond the core itself. This risk assessment has, therefore, focused on initiators which affect the ability to control radionuclides in the core itself.

The sixth and final level identifies the three functions which have been identified as critical to containing the activity in the MHTGR core, specifically within the fuel particle coatings. The first two are necessary to ensure the thermal limits of the particle coating are not exceeded. Heat generation must be controlled and adequate heat removal provided to prevent temperature conditions that could result in particle coating failure. Additionally, the particle coating must be protected against any chemical attack which could cause its failure. Subsequent sections focus on the potential threats of these three critical functions.

4.3 Initiators resulting from faults in plant systems

4.3.1 Initiators Challenging Heat Generation Control

The principal means of controlling heat generation is the control of neutrons in the core as is performed by the NCSS. The NCSS consists of a system of control rods used for normal power control and independent RSCE.

In general, it has been found that with regard to plant condition, the full power operating mode is limiting for the MHTGR. This is because the reliability of cooling systems is not significantly impacted by plant operating states. In the MHTGR, the same cooling systems may be employed whether the plant is pressurized or depressurized, for example. Furthermore, since the safety philosophy of the MHTGR has been to control radionuclides primarily within the fuel particle coatings, operating mode changes to secondary boundaries have less impact. Thus, the risk limiting condition has been found to be when the reactor is at power and core fuel and component temperatures are highest.

Alternatively, a condition arises wherein the heat generated by the core exceeds the heat removal. This can be from one of two causes:

(1) heat generation is not decreased when required by a decrease in heat removal; or (2) an undesired increase in heat generation occurs which exceeds the heat removal capability. Either has the potential to result in localized overheating of fuel particles and potential releases which require further consideration.

The events resulting in a decrease in heat removal without a consequential decrease in power generation might be initiated by an event which leads to a heat removal reduction followed by failure to reduce power or trip the reactor. Events which may lead to heat removal decrease are generally considered in the next section.

Failure to trip the reactor results if neither appropriate automatic nor manual action is taken. An automatic trip failure might be caused by instrumentation failure, control logic failure, or a mechanical control failure which prevents the insertion of an adequate quantity of material. Similarly, manual failure to trip could result from instrumentation failures, mechanical failures or operator failure. In assessing the likelihood of a successful manual trip, consideration must be given to the nature of the initiating transient. For example, the probability of the operator failing to trip manually during a rapidly occurring event may be high; while in slowly developing transients, the probability of operator error is lower.

The above events may generally be classified under the heading of Anticipated Transients Without Scram (ATWS) events. The severity of such events in the MHTGR is generally limited by a strong negative temperature coefficient which inherently serves to reduce heat generation.

This negative temperature coefficient, however, does not reduce power levels to normal decay heat levels and thus core and coolant temperatures above those encountered in decay heat removal scenarios will be encountered in the core. Thus, these events require further consideration in this assessment.

Other failure modes which might lead to undesired increases in heat generation (i.e., reactivity insertion events) that also require further consideration. Of these, the event which results in the largest reactivity insertion into the core is the control rod bank withdrawal event, the consequences of which would be generally worse under normal operating conditions. Based upon current control schemes for the MHTGR, a control rod group withdrawal would result in the removal of three outer control rod pairs with a combined reactivity worth well in excess of one dollar. Again, the severity of such a withdrawal is inherently limited by the strong negative temperature coefficient. However, core power level would exceed 100% causing core and coolant temperatures to exceed those encountered in typical decay heat removal scenarios. Therefore, further consideration of these events is required.

*4.3.2 **Initiators Challenging Heat Removal***

The principal means of heat removal in the MHTGR is by the Heat Transport System (HTS). The HTS consists of a single loop subsystem of a motor-driven helium circulator, a steam generator and associated feedwater and condensate system. The HTS removes heat from the core in normal operation as well as under shutdown conditions. An independent backup SCS is also provided in the design to remove core heat when the HTS is unavailable for maintenance or other reasons. The SCS consists of a single loop system with a motor driven helium circulator, a shutdown heat exchanger and associated pressurized water cooling system. The SCS is designed only for decay heat removal operation. A third system, the RCCS, is also provided for heat removal. Even should the RCCS fail, the function of heat removal from the core can still be maintained as heat is transported to the surrounding environment. While operation of the RCCS does reduce core temperatures somewhat in the event of the loss of all other cooling systems, its main function is to limit vessel temperature, thus ensuring vessel integrity. Additionally, the RCCS is totally passive with only structural components..

The failure modes of HTGR structures which should be considered for their impact on plant risks are leaks (i.e., the radionuclide barrier capability is compromised) or structural failures (i.e., other functions may be compromised).

Faults in the fuel particle coatings which allow the leakage of fission products into the primary coolant stream must be considered in the design of the MHTGR. The principal consequence of such faults is a pre-existing radionuclide source within the primary coolant system.

Table 4.2 Failure modes and their effects

Failure mode	Condition	Failure effect
Leaks in fuel particle coating Manufacturing defects Faults during operation	Power Startup/shutdown Shutdown Refueling	Release of fission products to primary system resulting in circulating and plateout activity sources.
Core structural fault Core support failure Core barrel failure Graphite block breakage	Power	Prevention of control rod and/or RSCE insertion. Reduction in cooling flow to core fuel. Reactivity change due to core configuration change
Structure: Primary coolant boundary components		
Helium leaks Pressure relief open Instrument line leaks Helium purification	Power Startup/shutdown shutdown	Depressurization of primary system through opening. Release of activity stored in the primary coolant system. Reduction in forced cooling effectiveness due to density decrease challenging heat removal function. Some air ingress challenging control of chemical attack
Water leaks System generator tube leak SCS heat exchanger leak Circulator auxiliaries leak	Power Startup/shutdown Shutdown Refueling	Depends on system and plant conditions Worst case is failure of steam generator tube at power resulting in water steam ingress into primary system. Water in coolant improves core moderation challenging the control of heat generatrion. Mass increase raises primary pressure leading to opening of primary reliefs. Moisture may cause hydrolysis of failed fuel and oxidation of core graphie threatening the control of chemical attack and generation of flammable gases
Vessel structural faults or breaks, Reactor vessel S/G vessel SCS closure Circulator closure Crossduct Support failure	Power Startup/shutdown Shutdown Refueling	Depressurization through a range of activity in the primary coolant system. Reduction in forced cooling effectiveness due to density decrease challenging heat removal function. Air ingress challenging control of chemical attack. If leak is larger than design basis, pressure forces may threaten other critical structures and components (i.e. reactor building, RCCS , control rods, circulators.)
Structure Primary coolant boundary components		
Reactivity control system structural fault NSSS closure failure causing rod ejection	Power Startup/shutdown	Similar to above. However, possibility of pressure forces ejecting control rod pair contained by housing challenging control of heat generation
Heat exchanger	Power	Similar to water leaks described above except

structural faults Steam generator tube sheet failure SCS heat exchanger tube sheet failure	Startup/shutdown Shutdown Refueling	greater ingress rate. If ingress rate exceeds design capacity of pressure reliefs, vessel integrity threatened.
Structure : Reactor building		
Leaks Overpressure louvers fail to close Closure failures	Power Startup/shutdown Refueling	Reduction in effectiveness of reactor building as a radionuclide barrier
Loss of structural integrity Missiles Pressure loads Seismic loads	Power Startup/shutdown Shutdown Refueling	Failure of reactor building as radionuclide barrier. Threat to heat removal function performed by RCCS in its structural integrity affected. No impact if other cooling systems continue to operate. Moderate impact on core temperatures if no other cooling systems operate, excessive vessel temperatures possible

A structural fault in the core or its associated support structures may be of concern because it can affect the functions of heat generation control and heat removal control. The challenge to the former function is likely more serious in the MHTGR. This is because the core has been designed such that only the heat transfer mechanisms of radiation and conduction are required to remove decay heat. Thus, even if a core geometry is altered such that coolant flow is impaired, consequences are minimal. In any case, the loss of forced convection cooling through the core is more likely to occur from cooling system failures than from core structural faults.

Geometry changes that might impact reactivity control systems are·those of greater concern, particularly if a geometry change are significant enough to affect the insertion of both control rods and the reserve shutdown poison. Because of the diversity of these two systems, either of which is capable of making the core subcritical, no internal mechanism has been identified with a likelihood high enough to be worthy of further consideration. However, as identified in Table 4.2, such structural faults do need to be considered from external initiators, particularly earthquakes.

As shown in Table 4.2, a structural fault in the primary coolant boundary components can present a potential threat to all three of the functions necessary to control radionuclides within the core fuel. In addition, such faults are of concern because they simultaneously cause a loss of the primary coolant system as a radionuclide barrier. Thus, considerable further analysis is warranted for these events. As shown in the table, two events in particular were selected for further analysis: a leak in the primary coolant boundary and a water ingress event caused by a failure in the steam generator.

With one exception, the selection of these two events is believed to adequately cover the range of failure modes possible to the primary coolant boundary. The exception, a fault in the NCCS vessel closure which might lead to rod ejection and a rapid reactivity insertion, is believed to be covered by the rod withdrawal event identified previously. This is justified, on the one hand, by the consideration that, based upon past reactor experience, the likelihood of a structural fault (i.e., a failure of a Class 1 pressure vessel component) leading to a rod ejection is orders of magnitude less than a control or operator fault leading to a control rod withdrawal.

On the other hand, the consequences of the rod ejection have been assessed to be only slightly greater than the rod withdrawal event. This may be attributed to a number of factors. First, the worth of a control group is greater than that of the rod pair which might be ejected. Second, the rod ejection is be limited by structures located above the rod enclosures. And finally, studies have shown the low power density fuel in the MHTGR experiences only a few hundred degree temperature rise before the

negative temperature coefficient terminates the event even for step reactivity insertions as large as the control rod pair which might be considered here. Thus, with the probability of the event being orders of magnitude lower and the potential consequences in terms of fuel behavior only slightly higher, the risks of the ejection accident are believed to be relatively insignificant.

Structural faults in the reactor building could minimize its effectivenss as a radionuclide barrier. This fault, however, if not coupled with another event which results in failures of the other radionuclide barriers (i.e., the core and the primary coolant boundary) would have virtually no consequence. A fault in the building could also impactthe effectiveness of the RCCS since it is supported off the building. Again, this fault, by itself, would have little consequences unless the other cooling systems failed (i.e., the HTS and SCS). The failure of the building combined with these other failures is believed to be extremely unlikely unless coupled by some event, such as a large earthquake. For this reason, reactor building failure was selected for consideration under the earthquake event tree identified in the following section.

*4.3.3 **External Initiators***

The candidate external events which have been considered for further evaluation in this risk assessment are identified in Table 4.3. Due to site specificity of external hazards and limitations in the availability of previous external event hazard assessments that are readily applicable to the MHTGR, the total safety risk impact from these events cannot be characterized without additional analysis exceeding the scope of this study. Other risk assessments, however, have shown that for a well designed plant (i.e., one whose risks are relatively low from internal faults), earthquakes may dominate the high consequence/low frequency end of the risk spectrum. Further, external events such as floods and fires generally only threaten the active safety systems, While large earthquakes pose a threat to key structures as well. Since the MHTGR has been designed to inherently minimize the consequence of active system faults, the structural threat from earthquakes to passive structures may be similarly limiting.

Table 4.3 External initiating events

Event	Remarks
Aircraft impact	Site specific; requires detailed study.
Avalanche	Can be excluded for most sites in the United States- and in Poland
Coastal erosion	Including in the effects of external flooding.
Drought	Excluded because ultimate heat sink is not affected by drought (e.g., air cooling).
External flooding	Site specific; requires detailed study.
Extreme winds and tornadoes	Site specific; requires detailed study.
Fire	Plant specific; requires detailed study.
Fog	Could increase the frequency of man-made hazard involving surface vehicles or air-craft; accident data include the effects of fog.
Forest fire	Fire cannot propagate to the site because the site is cleared; plant design and fire-protection provisions are adequate to mitigate the effects.
Frost	Snow and ice govern.
Hail	Other missiles govern.

High tide, high leak level, or high river stage	Included under external flooding.
High summer temperature	Ultimate heat sink is designed to operate with air
Hurricane	Included under external flooding; wind forces are covered under extreme winds and tornadoes.
Ice cover	Ice blockage of river included in flood. Potential loss of airflow passage is considered in plant design.
Industrial or military facility accident	Site specific; requires detailed study.
Internal flooding	Plant specific; requires detailed study.
Landslide	Can be excluded for most sites in the United States.
Lightning	Considered in plant design.
Low lake or river water level	Ultimate heat sink is designed for operation with air.
Low winter temperature	Thermal stresses and embrittlement are insignificant or covered by design codes and standards for plant design.
Meteorite	All sites have approximately the same frequency of occurrence.
Pipeline accident (gas,etc.)	Site specific; requires detailed study.
Intense precipitation	Included under external and internal flooding.
Release of chemicals in onsite storage	Plant specific; requires detailed study.
River diversion	Not applicable for air-cooling.
Sandstorm	Included under tornadoes and winds; potential blockage of air intakes with particulate matter is considered in plant design.
Seiche	Included under external flooding.
Seismic activity	Site specific; requires detailed study.
Snow	Plant designed for higher loading; snow melt causing river fooding is included under external flooding.
Soil shrink-swell consolidation	Site-suitability evaluation and site development for the plant are designed to preclude the effects of this hazard.
Storm surge	Included under external flooding.
Transportation accidents	Site specific; requires detailed study.
Tsunami	Included under external flooding and seismic events.
Toxic gas	Site specific; requires detailed study.
Turbine-generated missile	Plant specific; requires detailed study.
Volcanic activity	Can be excluded for most sites in the United States.
Waves	Included under external flooding.

The range of initiators selected pose challenges to the full range of safety functions and radionuclide barriers. Initiators involving anticipated transients without scram and inadvertent control rod withdrawal have significance because they jeopardize the ability to control heat generation. Loss of HTS cooling is included because of the challenge to core heat removal. All three of these events have the potential for initiating sequences Which could lead to incremental fission product release from the fuel. Loss of offsite power is selected as a representative support system fault because of its potential to initiate a loss of electrical power supplies which commonly impacts the systems providing heat generation and removal control.

Accidents initiated by primary coolant leaks are addressed because they involve primary coolant boundary failure With subsequent release of circulating activity and a fraction of plateout activity. Further, the loss of pressure affects the functions of heat removal and may challenge fuel particle coating integrity owing to chemical attack by any introduction of air. Steam generator leaks into the primary coolant system are considered in addition because of the potential for chemical attack of the fuel by hydrolysis as well as having a positive reactivity effect which challenges control of heat generation.

Finally, earthquakes merit attention because they can cause normally independent plant systems or, potentially more important to the MHTGR, passive plant structures to fail concurrently. Thus, their ability to initiate common mode failures presents a potential challenge to all functions as well as all physical radionuclide barriers.

4.3.4 ***Results.***

Table 4.3 Summary of accident initiators selected for further analysis

Initiating event	Function challenged			Barriers challenged		
	Heat generation	Heat removal	Chemical attack	Fuel particle	Primary coolant	Reactor building
Critical system faults						
Anticipated transient without scram	x			x		
Control rod withdrawal	X			X		
Loss of heat transfer system		x		x		
Support system faults						
Loss of offsite power	X	X		X		
Critical structure faults						
Primary coolant system leaks		x	x	x	x	X
Steam generator leaks	X		X	x	X	
External faults						
Earthquakes	X	X	X	X	X	X

4.3.5 ***Plant response***

For each transient described in this section, the response of the plant protection and instrumentation system (PPIS) to the transient under consideration is provided. The PPIS receives the actuation signals from its sensors and responds accordingly. The PPIS initiates reactor trip, heat transport system (HTS) shutdown, shutdown cooling system (SCS) initiation, steam generator isolation and dump, and primary coolant pumpdown as necessary..

4.3.6 ***Primary Coolant Leaks***

Failure of the primary coolant pressure boundary to retain the helium inventory may result from a variety of causes. Included as possible failure locations are instrument lines, neutron control system guide tubes, valves, vessel penetrations, flanged or bolted closures, and welds. Response of the plant

varies depending upon the size of the leak area and the consequent depressurization rate of helium from the reactor vessel.

4.3.6.1 Planned Plant Response.

The expected plant response to a primary coolant leak begins with reactor trip with the control rods. This trip is initiated by the PPIS when reactor vessel pressure is reduced to 5688 kPa. The helium purification system (HPS) pump down function is started on a signal from the PPIS when a primary coolant pressure of 5515 kPa is reached and high reactor building radiation levels are detected. This action is ineffective when the leak size is large enough, since the pumpdown rate is negligible compared to the depressurization rate of the reactor vessel through the breach in the pressure boundary. Core cooling continues on the HTS for leak sizes smaller than 0.65 cm^2. For larger leak sizes, HTS trip is initiated by the PPIS upon detection of low primary system pressure [(4412 kPa «640 psia)] and a relatively high steam temperature [)393°C ()739°F)]. HTS shutdown in turn signals an SCS startup following which the SCS serves to remove the core decay heat. Although the PPIS trips the HTS in response to 0.65 cm 2 (0.1 in.2) and larger primary coolant leaks, decay heat removal with the HTS can be resumed within an hour. Thus, the normal response to 0.65 cm2 (0.1 in.2) and larger primary coolant leaks is reactor trip~ HPS pumpdown initiation, HTS trip, SCS initiation, and eventual resumption (manually) of HTS cooling following an interval of decay heat removal with the SCS.

4.3.6.2 Plant Response to Abnormal Conditions.

Failure to provide forced core cooling following a primary coolant leak or failure to pumpdown the primary coolant inventory to storage may occur. Depending upon the leak location and size, HTS or SCS failure may result from damage directly attributable to the depressurization. Otherwise, failure of these systems is independent of the initiating event.

Another consequence of primary coolant leaks which must be considered is the possible entry of air into the primary system and the potential for air-graphite reaction. An analysis was made of the fraction of graphite reacted as a function of time following a 33 cm^2 leak (the flow area of the largest pipe, relief valve line, connected to the vessel). In such an event the air ingress is relatively limited, occurring as a result of hydrostatic displacement and thermal contraction. The amount of graphite oxidized in such an event is small (0.01%) with no significant threat to core integrity. The quantity of flammable gas generated in such an event (CO) and released to the reactor building through the leak area has also been calculated and is well below flammability or explosive limits.

For still larger though extremely unlikely leak areas (i.e., those involving a failure in the Class 1 vessel or vessel closures), there is the concern of pressure induced failure of critical structures. Large pressure forces might, for example, cause disruptions in the core geometry convection cooling pathways or neutron control poison pathways.

However, gross changes in the convection cooling pathway would be no more significant than the loss of forced cooling events considered in the next section and clearly less likely. Similarly, alterations in the poison pathway would be no more serious than the ATWS events.

Another possible concern, is that overpressurization of the reactor building and failure of the RCCS cooling panels might occur.

There is also concern that in the event of a large leak, greater core oxidation might occur. The air flow which can access the core is generally limited through any single hole. The worst case may be considered to be the hypothetical simultaneous occurrence of holes at the bottom and top of the vessel which might allow air to freely convect through the core. Even in such an event the amount of air which could DOE-HTGR-86-011/Rev. 3 react with the core would be limited by the air in the building volume and subgrounded reactor cavity. Analysis of the reaction of the complete reactor building volume of air with th core graphite indicates that less than 1% of the graphite would be reacted, and no significant threat to fuel integrity would be expected.

If the reactor building and subgrounded core cavity are postulated to be ineffective in limiting air ingress, the graphite reaction would be limited only by the pressure drop across the core. Because the graphite-air reaction is exothermic there is the concern that the heat added by the reaction might accelerate the fuel heatup and increase the ultimate temperature leading to significant incremental fuel failures.

Bounding analyses of this event have therefore been conducted to determine the additional heat added owing to the graphite reaction.

Such analysis indicates the exothermic energy added by this reaction would only be a few percent of decay heat levels after plant shutdown.

As such, core heatup would remain slow with peak temperatures and incremental fuel fission product release not occurring until days after the event, allowing ample time to take mitigating actions before such actions would be hampered by high radiation levels.

*4.3.7 **Loss of the Heat Transport System***

Loss of the HTS may be engendered by failures either within the BOP or nuclear steam supply system (NSSS) portions of the plant. BOP failures may eventually result in a loss of feedwater flow to all four modules. NSSS failures should only impact one module.

4.3.7.1 Planned Plant Response.

The planned response to a loss of HTS cooling caused by a secondary cooling system failure is to disengage the circulator motor contacts and close the steam generator steam and feedwater block valves. Complete HTS shutdown is accomplished following detection by the PPIS of a circulator speed to feedwater flow mismatch (i.e., ratio greater than 1.20). Following HTS shutdown, the reactor is tripped with the control rods. This is accomplished through detection of two separate trip parameters. The safety protection subsystem (SPS) initiates a reactor trip upon sensing a high neutron flux to helium mass flow ratio. The Investment Protection Subsystem initiates reactor trip on a redundant trip signal (i.e., HTS shutdown) and also initiates a startup of the SCS for each module that has lost HTS cooling. In the event that the outer control rods are not inserted into the reactor core, a second trip setpoint for RSCE actuation is reached when the neutron flux to HTS circulator speed ratio reaches 1.80 and a 30 s time delay has passed since signaling the outer control rods to drop. In this manner, redundancy is achieved in the ability to control neutron flux levels in the reactor core.

Failures within the NSSS portion of the plant include spurious circulator trip and other localized failures of the HTS. Identical reactor trip signals as indicated above apply to this type of failure as well, except only one module is affected. The SCS is again started automatically for the module that has lost HTS cooling.

4.3.7.2 Plant Response to Abnormal Conditions.

Loss of HTS cooling may be followed by the failure of the SCS to start or run. Cooldown of the reactor core is then accomplished by conduction and radiation to the RCCS cooling panels. The vessel pressure rises initially but remains low enough to avoid lifting the primary relief valves [which are nominally set to open at 7177 kPa (1041 psia)] thus containing fission products within the reactor vessel. The primary coolant pressure transient corresponds to a pressurized cooldown on the RCCS, without any HTS or SCS decay heat removal.

*4.3.8 **Earthquakes***

Plant response to an earthquake may vary depending upon the seismic intensity range under consideration. Ground accelerations below 0.2 g are not expected to result in system failures. At intensities below an operating basis earthquake (OBE) (0.15 g for the MHTGR) reactor trip may not be

required and normal power production can be maintained. For large earthquakes above a 0.4 g ground acceleration, forced cooling systems may be lost and plant damage may occur.

4.3.8.1 Planned Plant Response.

The initial response of the plant to large earthquakes, on the order of the 0.3 g safe shutdown earthquake (SSE), is an HTS trip of all modules. The cause of the trip may arise from a variety of disturbances which lead to abnormalities in the feedwater or helium flow. Seismically induced electrical faults are generally responsible for these disturbances during such an event. HTS trip is typically initiated following detection of the circulator speed to feedwater flow mismatch created by the disturbance. The reactors are tripped automatically by the PPIS following the signal to trip the HTS in all modules. The main circulators coast down in approximately 2 min, at which time the helium shutoff valve closes by gravity due to decreased helium flow. Following reactor trip, decay heat levels are quickly reached and the reactor pressure remains essentially constant, rising only slightly, until core heat removal is resumed by the SCS.
The main loop trip signal initiates cooling on the SCS in all four modules 400 s after the transient begins. Following SCS startup, the pressure in the primary system begins to decrease.

4.3.8.2 Plant Response to Abnormal Conditions.

Following a large earthquake and HTS trip, heat removal capabilities may be challenged by failure of the SCS in one or more modules. The response of the plant in such an event is to remove heat from the core by conduction and radiation to the RCCS cooling panels.

In the extremely remote event of earthquakes occurring much larger than the design basis SSE event (i.e., earthquakes with accelerations much larger than 0.3 g), critical structures may be threatened. In particular, failure of the passive RCCS may need to be considered in addition to failures of the active forced cooling systems. It should be recognized that the RCCS is designed with considerable margin and redundancy. Thus the earthquake would have to cause multiple structural failures and such failures would have to be virtually total in terms of cutting off the air flow to the vessel cooling panels.

In such a remote event, core heat would be removed only by the mechanisms of radiation and conduction to the structures and earth surrounding the reactor cavity. Natural convection currents in the silo cavities also aid heat removal. Analyses have been conducted to determine fuel temperature response in this case with no heat removal via the RCCS. In such a case, peak and average fuel temperatures exceed those which are calculated when the RCCS is functioning by some several hundred degrees. The core maximum and average temperatures reached are approximately 1870 °C (3398 °F) and 1600 C (2912 °F). At these higher fuel temperatures, incremental fission product release from the core does 0occur. As identified in Section 8, in fact, the activity release from this event involving all the four reactor modules (designated DC-1), bounds all other events assessed in this risk assessment. However, even with these degraded heat removal conditions, the inherent thermal response of the core and fuel particle retention are still adequate to limit releases to a fraction of a percent (0.02% of halogens for example) and thus prevent any truly gross release of core fission products.

4.3.9 Loss of Offsite Power

MHTGR electrical loads are supplied by the house turbine-generator sets and offsite power sources. If offsite power is lost, the house electrical loads are supplied by the turbine-generators. Either turbine-generator set is capable of sustaining house loads.

Loss of offsite power accompanied by an inadvertent turbine trip results in a sustained loss of all nonuninterruptible ac power. Electrical systems which still remain available to serve vital components are the uninterruptible power supplies, dc battery power systems, and the backup generators. DC battery power is available for up to 1 h at rated load to supply the uninterruptible power sources until standby generators are started.

4.3.9.1 Planned Plant Response.

The MHTGR control system is similar in design to the British gas-cooled reactors insofar as both are designed to remain online in the event of loss of offsite power. The probability is high that at least one of the two turbine-generators will remain online and provide power to in-house electrical loads following the loss of offsite power.

4.3.9.2 Plant Response to Abnormal Conditions.

The plant response to a loss of offsite power and inadvertent trip of both ECS turbines is a concomitant trip of the HTS circulators in all four modules and of the feedwater pumps, due to the loss of ac power to these components. Power continues to be supplied to the PPIS through the uninterruptible power source which is supplied by the dc power system batteries. HTS shutdown is signaled by the PPIS upon detecting a circulator speed to feedwater flow mismatch. The signal to shut down the HTS results in (1) the main steam and feedwater block valves being closed, (2) a signal to trip the reactor, and (3) a signal to initiate the SCS for decay heat removal.

Reactor trip may also be initiated by detection of a high neutron flux to helium mass flow ratio. If the control rods are not dropped, reserve shutdown material is inserted into the reactor core within 30 s, thus providing redundant and diverse capability for reactor trip. Power to the SCS to provide decay heat removal is accomplished by relying on the backup generator sets. Since SCS cooling is provided, there is no challenge to retention of radionuclides in the fuel. In addition, the resulting pressure transient while on the SCS does not challenge radionuclide retention by the reactor vessel.

If the backup generators fail to power SCS cooling, this results in a loss of all forced cooling mechanisms. Core decay heat is then removed by conduction and radiation of heat to the RCCS. In this abnormal plant condition, the pressure transient does not challenge radio-nuclide retention by the reactor vessel. Figure 6-3 again provides the resultant pressure transient.

Plant Response to Abnormal Conditions.

Failure to provide HTS cooling is the most likely abnormal response to a control rod bank withdrawal.

An analysis has been conducted of a hypothetical withdrawal of all control rods (worth 3% ~k) at full power without scram and loss of all forced core cooling. Core power increases until the negative worth of the resulting temperature increase in the fuel balances the positive worth associated with the rod withdrawal. The fuel, in such a case, heats up, but stabilizes at a temperature well below the point at which experimental evidence indicates there would be any significant incremental failures in the fuel particle coatings.

4.4 System reliability models

Maintaining the ability to adequately remove decay heat is of concern in all event sequences evaluated in subsequent sections. Included in the evaluation of sequences where all cooling systems are lost, is the ability to repair at least one of those systems before important safety limits are exceeded. The component found to be most sensitive to excessive thermal transients was the reactor vessel. Under pressurized conditions, the vessel is conservatively assumed to fail if it exceeds 482 °C. Under depressurized conditions, the vessel material undergoes a phase change at 760 °C. The mission time for the HTS, SCS, and RCCS cooling systems under depressurized and pressurized accident conditions was evaluated and used to construct Fig. 6-18 which depicts maximum time to repair a cooling system versus the prior cooling time. The mission time is defined as the time any cooling system must operate (HTS, SCS, or RCCS) to prevent vessel failure resulting from exposure to excessive temperatures.

4.4.1 HTS Cooling

The primary means by which heat is removed from a module during both normal operation and shutdown conditions is by the HTS. Heat is transferred to the circulating helium in the core which then

is routed through the steam generator. As the helium flows through the steam generator, heat is transferred through the steam generator tubes to the secondary side water.

Failure of both ECS trains engenders the loss of HTS cooling in all four plant modules. The ECS is the heat sink for the HTS.

4.4.2 ***SCS Cooling***

In the event the HTS fails to remove decay heat in one or more modules, the alternative is decay heat removal by the SCS. The SCS is normally in a standby mode, and is signaled to start by the PPIS once the HTS has been lost for any reason. In some cases, the SCS reliability is degraded if HTS failure was a result of failure of a support system common to both the HTS and SCS. Successful operation of the SCS results in a redundant means of decay heat removal following plant shutdown.

The SCS may be required to operate in more than one module if the HTS failure was a result of the failure of systems Which support the HTS in all modules, such as the plant service water subsystem, RPCWS, TBCCWS, or electrical systems.

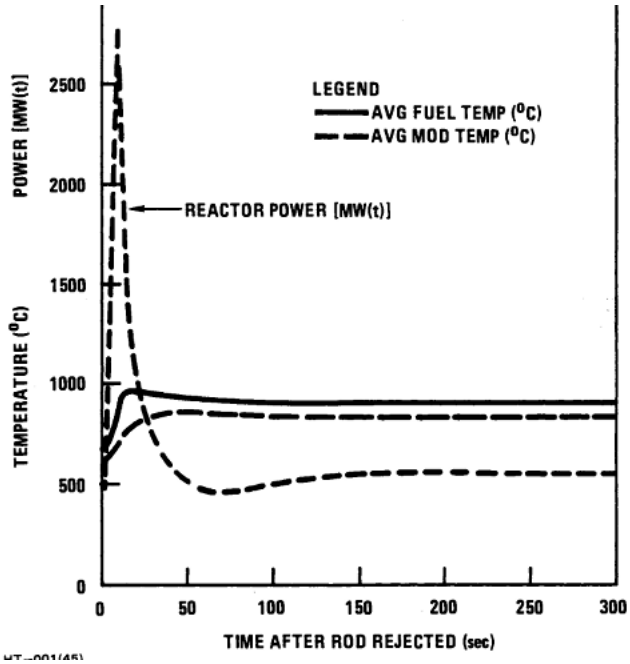


Fig. 4.1. Hypothetical control rod ejection at power without scram

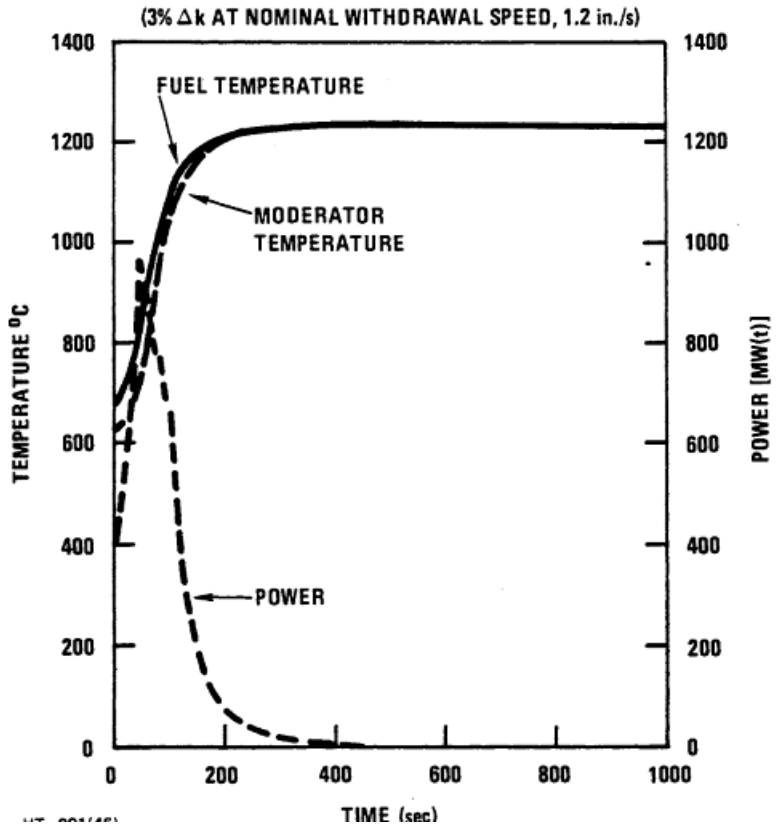


Fig. 4.2. Hypothetical removal of all control rods at full power without scram

4.5 Frequency

The primary focus in the MHTGR safety approach is on the retentive properties of the ceramic, high-temperature fuel. The events considered in this assessment present a spectrum of challenges to this design goal of maintaining control of radionuclide release through retention by the fuel. These challenges can be deemed as being associated with one of three major manners in which the fuel retention can be compromised: failure to control heat generation; failure to maintain core heat removal; and failure to prevent chemical attack. Thus, beyond the release of circulating activity involved in a primary coolant leak, the real threat from such a breach of the primary coolant boundary is the potential for chemical attack of the core.

*4.5.1 **Primary coolant leaks***

As an initiating event, primary coolant leaks are of interest for several reasons. Because of the activity circulating with the primary coolant or plated out around the primary coolant circuit, failure of the primary coolant pressure boundary necessarily results in some, albeit limited, release of radionuclides to the environment regardless of any subsequent plant response. Additionally, if the leak is of sufficient size, the damage to surrounding equipment resulting from the leak may threaten the integrity of core cooling systems, and allow for graphite oxidation as a result of air ingress. Given that a leak occurs, various possible plant responses which affect consequence determination are possible. The assessment shows primary coolant leaks to be the most likely reason of radionuclide release. Very small leaks (<0.65 cm^2) are predicted to occur more often than once in four years of plant operation. Larger leaks, up to failures of the relief valve train line, are shown to be significantly less likely. Further, the assessment shows the likelihood of a loss of forced circulation cooling occurring in combination with a primary coolant leak to be approximately twice in a thousand years of plant operation (2×10^{-3} per plant year).

*4.5.2 **Loss of main loop cooling***

The loss of main loop cooling is initiated by equipment failures within the plant which preclude continued operation of the HTS in one or more modules. As an initiating event, the loss of main loop forced circulation core cooling is of interest as a challenge to the function of removing core heat and consequently a potential precursor to the incremental releases from fuel.

Given that such an event occurs, various possible MBTGR responses resulting in differing alternative cooling modes are possible. The assessment shows the likelihood of event sequences that could lead to a radionuclide release as a consequence of a loss of main loop cooling to be extremely remote. The mean frequency of any such sequence has been assessed at just under 1 x 10-7.

*4.5.3 **Earthquakes***

The equipment damage produced by the vibrations during an earthquake causes seismic events to be the most important class of external events because it (1) simultaneously challenges redundant equipment in each of the modules; and (2) poses one of the few potential risks to passive equipment. The radiological risk from seismic events is nevertheless limited because severe earthquakes with intensities sufficient to damage key systems and structures are very unlikely, and only a few components are required to function in any case. No event sequence with a radionuclide release is predicted to occur with a mean frequency of greater than 7×10^{-7} per year, at which point earthquakes of sufficient severity to damage the primary coolant boundary are predicted.

*4.5.4 **Loss of offsite power***

The normal station electrical power equipment refers to the normal loads in the energy conversion train for power production such as the HTS circulators, condensate pumps, and feed pumps. A loss of normal station power (LOSP) occurs when, for any reason, the power flow from the grid (via the main or auxiliary transformers) is lost and the turbine generators inadvertently trip instead of maintaining their load and continuing to remove heat.

A LOSP is of interest as an initiating event because it is externally caused, and because it can simultaneously challenge multiple systems. For example, if offsite power is lost and both turbines trip, main cooling loops in all four modules are shut down which challenges core heat removal and, consequently, may result in incremental fuel releases from thermal mechanisms.

Nevertheless, the passive features at the MHTGR are such that no event sequence which could result in radionuclide release is predicted to occur within the frequency range considered in this study.

*4.5.5 **Anticipated transients requiring scram***

There are a number of off-normal plant transients for which the PPIS is designed to detect the upset condition and as a part of the automatic response, reduce the heat that must be removed from the core by initiating a reactor shutdown (scram) in one or more modules with the control rods. Such a transient without successful scram is of interest as a challenge to the continued control of core heat generation. In challenging this function, the ATWS represents a potential precursor to failure of the primary coolant boundary (relief valve lifting) simultaneous with the incremental releases from fuel involving thermal effects. However, the negative temperature coefficient and high temperature integrity of the ceramic core provide the MHTGR with the capability to sustain such an ATWS for extended periods of time without adverse consequence. This, when considered with the high reliability of two diverse shutdown systems, results in the assessment showing no event sequences with radiological release within the range of frequencies studied.

*4.5.6 **Inadvertent control rod withdrawal***

Inadvertent control rod withdrawal is initiated by failures in the rod control equipment that lead to the undesired withdrawal of one or more control rods from the core. As an accident initiating event, rod withdrawal is of interest because of its potential challenge to the continued control of core heat

generation. In challenging this function, the rod withdrawal represents a potential precursor to failure of the primary coolant boundary (relief valve lifting) simultaneous with the incremental releases from fuel involving thermal effects. However, withdrawal of a complete control rod group results in only localized fuel temperature rise and by itself is not expected to result in any offsite consequence. This has been shown to be true even in the case where reactor trip fails to occur.

As a result, the frequency assessment for this accident initiator has shown no event sequence of meaningful probability initiated by control rod withdrawal and having the potential to result in offsite release.

*4.5.7 **Steam generator leaks***

Water ingress is selected as an initiating event because of the potential for primary coolant release due to relief valve venting and for incremental fuel releases due to chemical attack (hydrolysis) of the fuel. Furthermore, water ingress is of interest due to its reactivity effect on the core. In the assessment it is shown that the most likely event sequences following a moisture ingress result in no dose. Only in those sequences where certain mitigating features (e.g., steam generator isolation and dump) fail to perform their functions, are offsite doses predicted. The likelihood of such a water ingress induced release is assessed at somewhat less than once in 10,000 yr (1×10^{-4}/ year). Further the assessment shows the likelihood of a water ingress induced release in combination with a failure of forced circulation core cooling to be approximately four times in 100,000 yr of plant operation (4×10^{-5} per plant year).

4.6 Effects

*4.6.1 **Consequences from forced convection cooldowns under dry conditions***

A number of release categories that are initiated by primary coolant leaks have been identified as forced convection cooldown under dry conditions. The categories are labeled DF-1 through DF-4 where DF-1 has the greatest consequence and DF-4 has the least nonzero consequence. The categories are described in Table 4.3. The consequence source term for forced convection cooldowns under dry conditions includes a portion of the circulating activity and the liftoff of a fraction of the activity plated-out on primary circuit surfaces. Incremental release of radionuclides from the fuel body inventory is prevented by forced convection cooling of the reactor core, which is provided in all cases by either the HTS or the SCS.

Table 4.3 Release Category Descriptions For Forced Convection Cooldowns Under Dry Conditions

Primary coolant leak occurs where 6.5 cm2 Area < 84 cm2
Reactor is tripped.
BTS or SCS maintains forced convection cooling.
BPS pump down is ineffective.
Radionuclides are released through the reactor building dampers.

The circulating and liftoff activities are released through the breach in the primary coolant boundary into the reactor building. For smaller leak sizes, the consequences are reduced by pumpdown of primary coolant to storage bottles by the BPS. For larger leak sizes pump down becomes ineffective, and essentially 100% of the circulating activity is released into the reactor building. The fraction of material lifted-off at a given location in the primary circuit increases when helium flow velocities increase at the location. Once in the reactor building, fission products are depleted by the natural processes of radioactive decay, plateout on building surfaces, and by particulate settling. The fission products can be transported from the reactor building to the atmosphere by building leakage or

through the building dampers if the depressurization rate from the vessel exceeds the building leak rate.

Table 4.4 The median, ninety-fifth percentile, and fifth percentile results of the dose uncertainty analysis for forced convection cooldown under dry conditions in HTGR

Release category	Leak size cm2	Dose in mSv					
		Whole body, gamma			Thyroid		
		5%	Median	95%	5%	Median	95%
DF-1	6.5	4,5E-4	2,9E-3	2,7E-2	2,5E-3	1,4E-2	9,1E-2
DF-2	0.65	2.,2E-4	1,4E-3	1,1E-2	2,6E-4	3,5E-3	4,5E-2
DF-3	0.065	2,6E-5	1,7E-4	1,8E-3	4,6E-5	6,6E-4	8,5E-3
DF-4	0.0065	1,5E-5	9,2E-5	9.0E-4	1,4E-5	2,5E-4	3,0E-3

Release category descriptions for forced convection cooldowns under wet conditions:

- Moderate steam generator leak occurs.
- Reactor trip occurs.
- Steam generator isolation from steam heater is delayed.
- Isolation and dump of the steam generator occurs within 20 min.
- SCS maintains forced convection cooling.
- Primary relief valve opens and fails to reclose.
- Radionuclides are released through the reactor building dampers.

In all cases, the reactor core is cooled by forced convection provided by the SCS, which prevents any thermally induced incremental release of radionuclides from the fuel body inventory.

*4.6.2 **Consequences from conduction cooldowns under dry conditions***

A number of event sequences that are initiated by primary coolant leaks and seismic activity have been grouped and categorized as conduction cooldowns under dry conditions.

Table 4.5 The median, ninety-fifth percentile, and fifth percentile results of the dose uncertainty analysis for conduction cooldowns under dry conditions in HTGR

Release category	Dose at EAB during 30 days in mSv					
	Whole body, gamma			Thyroid		
	5%	Median	95%	5%	Median	95%
DC-1	1,0E-1	6,4E-1	4,5	6,0E+1	5,0E+2	4,3E+3
DC-2	2,7E-2	1,6E-1	1,1	3,0E+1	2,5E+2	2,1E+3
DC-3	4,8E-4	2,9E-3	2,3E-2	2,0E-1	2,1E0	2,2E+1

DC-4	1,5E-4	1,5E-3	1,5E-2	7,6E-2	7,6E-1	7,6E0

The consequence source term for conduction cooldowns under dry conditions includes (1) the circulating activity, (2) fission product release from the fuel due to high temperatures, and (3) liftoff of a portion of the activity plated-out on primary circuit surfaces.

The median, ninety-fifth percentile, and fifth percentile results of the dose uncertainty analysis for thyroid and whole body gamma doses from conduction cooldowns under dry conditions for a 30-day exposure at the EAB are presented in Table 4.5.

Table 4.6 The median, ninety-fifth percentile, and fifth percentile results of the dose uncertainty analysis for forced convection cooldowns under wet conditions in HTGR

Release category	Dose at exclusion area boundary in mSv					
	Whole body, gamma			Thyroid		
	5%	Median	95%	5%	Median	95%
WF-1	2,6E-3	2,2E-2	1,9E-1	3,8E-1	3,4E0	3,1E+1
WF-2	2,0E-3	1,7E-2	1,4E-1	3,1E-1	2,8E0	2,5E+1
WF-3	3,9E-4	3,3E-3	2,8E-2	5,8E-2	5,2E-1	4,6E0
WF-4	4,8E-4	2,6E-3	2,2E-3	4,7E-2	4,2E-1	3,8E0

4.7 Risk assessment results

The results of this safety risk assessment show the MHTGR to behave in an extraordinarily benign manner with only limited offsite releases predicted during even extremely unlikely accidents. Accordingly, the concept is shown to comply with the risk limits of the NRC Safety Goals and to do so with substantial margin. The MHTGR is also shown to be capable of satisfying the very stringent user-imposed requirement that Protective Action Guideline (PAG) doses related to public evacuation and sheltering are met at the 425 m site Exclusion Area Boundary (EAB). PRA results demonstrate that releases with frequencies as low as 5 x 10-7 per year are below the PAG sheltering limits of 1 Rem Whole Body and 5 Rem Thyroid at the site EAB.

*4.7.1 **Quantification of risk***

The safety risk for the MBTGR is a function of the frequency of occurrence and the consequence to the public of the various accidents examined in this assessment. Quantification of that risk is performed by combining the accident frequencies and their uncertainties.

*4.7.2 **Mean Risk Estimate***

Mean risk is defined as the product of mean frequency and consequence. One method of interpreting the risk assessment results is to find this product for each of the different accident categories considered, sum the individual risks, and view the result as a single expectation value for plant risk. This approach, while having certain limitations, is simple and necessary if the plant is to be compared against risk limits that are included in the NRC Safety Goals. Such a treatment of mean risk has been done for the MHTGR in tabular form and is described here.

Table 4.7 shows the values for mean frequency and consequence for each of the categories release. As can be seen, the consequence for every category includes both a whole body and a thyroid exposure expressed in mSv. These calculated doses may be thought of as the expected doses to the

maximum exposed individual remaining at the plant EAB without evacuation. For each of the release categories, the frequency and consequence values have been combined, giving the mean risk to that theoretical individual, in mSv per year.

The mean risk represents the individual risk at the EAB from each of the accident categories. The total mean risk estimate is 3 x 10^{-4}/year whole body and 3 x 10^{-3} mSv per year thyroid. The release category dose and risk estimates in Table 4.5 are conservative relative to compliance with NRC Safety Goal and not directly comparable to other risk assessments since they were evaluated for an individual remaining at the plant EAB of 425 m for 30 days and nights. These are the conditions corresponding roughly to Polish dose limits, which are 10 mSv whole body dose over 12 months after the accident.

Table 4.5 Public risk from release categories

Release category	Mean frequency of release category (per year)	Mean dose (a) mSV		Mean risk (mSv/year)	
		Whole body	Thyroid	Whole body	Thyroid
DF-1	1E-2	1E-2	3E-2	1E-4	3E-4
DF-2	4E-2	3E-3	1E-2	1E-4	4E-4
DF-3	1E-2	1E-3	2E-3	1E-5	2E-5
DF-4	0,3	4E-4	9E-4	1E-4	3E-4
WF-1	5E-8	5E-2	8	3E-9	4E-7
WF-2	4E-6	4E-2	7	2E-7	3E-6
WF-3	1E-6	8E-3	1	8E-9	1E-6
WF-4	6E-5	6E-3	1	4E-7	6E-5
DC-1	9E-8	1	1E+3	9E-8	9E-5
DC-2	8E-8	0,3	5E+2	2E-8	4E-5
DC-3	7E-7	8E-3	5	6E-9	4E-6
DC-4	5E-5	4E-3	2	2E-7	1E-4
DC-5	3E-5	2E-3	1	6E-8	3E-5
DC-6	5E-4	2E-3	1	1E-6	5E-4
DC-7	3E-4	8E-3	1	2E-6	3E-4
DC-8	8E-5	8E-3	1	6E-7	8E-5
DC-9	2E-3	2E-4	0,2	4E-7	4E-4
WC-1	2E-7	2E-1	90	4E-8	2E-5
WC-2	3E-7	4E-2	10	1E-8	3E-6

WC-3	6E-6	7E-3	4	4E-8	2E-5
WC-4	2E-7	4E-3	2	8E-10	4E-7
WC-6	8E-6	1E-3	0,6	8E-9	5E-6
WC-7	4E-5	1E-3	9E-2	4E-8	4E-6
			Total risk	3E-5	3E-3

a) Doses and risk values evaluated for a maximum exposed individual located at the plant EAB of 425 m without evacuation.

The accident type made up of the several release categories prefixed DF is a transient where forced convection cooling is maintained with dry primary coolant conditions. This accident type includes events initiated by primary coolant leaks. The accident type made up of the several WF release categories is a transient where forced convection cooling is also maintained but where moisture ingress has led to K_{wet} primary coolant conditions. Category WF includes events initiated by small and moderate steam generator leaks. The accident types comprised of DC and WE release categories are conduction cooldown events in which all forced convection cooling provided by the HTS and the SCS is lost. Release categories designated DC are conduction cooldowns under dry conditions and include events initiated by primary coolant leaks, loss of main loop cooling, and seismic activity. Release categories prefixed with we are conduction under wet conditions and include events initiated by small and moderate steam generator leaks.

Referring again to Table 4.5, it can be seen that the highest whole body risks are from release categories designated DF. These categories are initiated by leaks in the primary coolant boundary with a resulting release of circulating activity and reentrained material normally plated out around the primary coolant circuit. Because forced cooling is maintained, there is no incremental fuel release. Note that even at the low frequency, higher consequence end of these release categories, EAB whole body doses in excess of 10 mSv are not predicted. None of the DF release categories has a particularly large dose associated with it. Rather, the importance of the DF categories to total risk stems from their relatively high frequency as compared to other release categories.

Not only are accidents belonging to category DF the largest whole body risk contributors, but also that they dominate the risk envelope for whole body gamma doses across the range of frequencies evaluated. This risk dominance occurs inspite of the low consequencies associated with this accident category

5 Fission product releases from MPBR

5.1 Containment of fission products in pebble bed MPBR

5.1.1 ***Radionuclide release barriers***

HTGR designs employ multiple radionuclide (RN) release barriers to meet RN control requirements

• RN transport in HTGRs has been extensively investigated

• Design methods are available to predict performance of the RN release barriers during normal operation and accidents release barriers during normal operation and accidents

– Codes have been used extensively for reactor design &analysis, including operating HTGRs

Release Barriers include:

– Fuel kernels

– Particle coatings (most important barrier)

– Fuel-element matrix/fuel-element graphite (prismatic)

– Primary coolant pressure boundary

– Reactor building (RB)

• These multiple RN barriers provide Defense-in-Depth

The sources of fission product releases and pathways are shown in the figure 5.1 below.

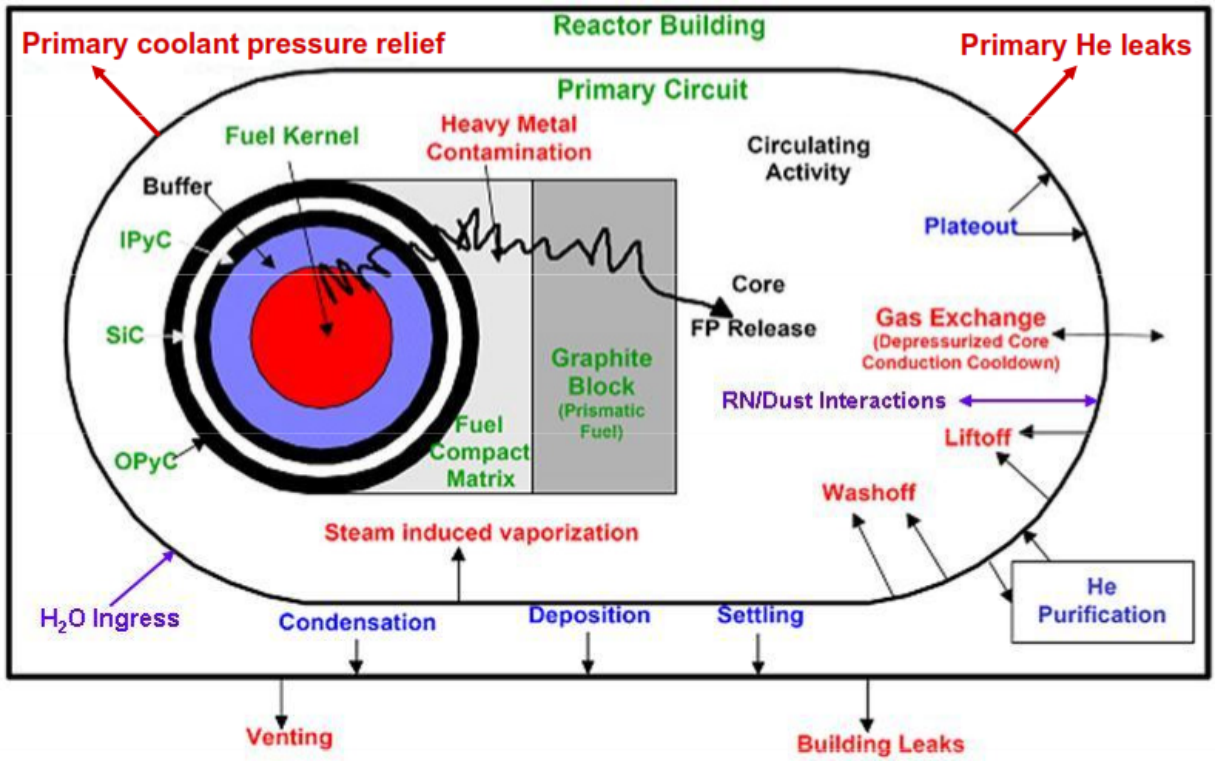


Fig. 5.1 Schematic of HTGR radionuclide sources and pathways (drawn from HTGR Technology Course module 13[36])

Green = barriers, red – releases, blue = removal from atmosphere, black – comments, violet – interactions

The effectiveness of the individual barriers in containing radionuclides depends upon a number of fundamental factors, including the chemistry and half-lives of the various radionuclides, the service conditions, and irradiation effects. The effectiveness of these release barriers is also event specific; for example, the attenuation of iodine release to the environment by the reactor building is much higher during core conduction cooldown events than during rapid depressurization events.

[36] HTGR Technology Course for the Nuclear Regulatory Commission, May 24-27, 2010, Module 13, Fission product behavior in HTGRs, Idaho National Laboratory, General Atomics.

The fourth release barrier in the HTGR is the primary coolant pressure boundary.

*5.1.2 **Fission products in the primary circuit***

Once the fission products have been released from the core into the helium coolant, they are transported throughout the primary circuit. The helium purification system removes both gaseous and metallic fission products from the primary coolant at a rate determined by the gas flow through the purification system. However, for the condensible fission products, the dominant removal mechanism is deposition, or "plateout", on the various helium-wetted surfaces in the primary circuit. The plateout rate is determined by the mass transfer rates from the coolant to the fixed surfaces, by the sorptivities of volatile fission products on the various materials of construction, and by the temperature of operation.

The circulating and plateout activities in the primary coolant circuit are potential sources of release into the environment in the event of primary coolant leaks or as a result of the venting of primary coolant in response to overpressuring the primary circuit, e.g., in response to significant water ingress. The fraction of the circulating activity lost during such events is essentially the same as the fraction of the primary coolant that is released, although the radionuclide release can be mitigated by pumpdown through the helium purification system if the leak rate is sufficiently slow.

A small fraction of the plateout may also be reentrained, or 'lifted off", if the rate of depressurization is sufficiently large. The amount of fission product liftoff is expected to be strongly influenced by the amount of paniculate matter in the primary circuit as well as by the presence of friable surface films on primary circuit components which could possibly spall off during a rapid depressurization.

Other mechanisms which can potentially result in the removal and subsequent environmental release of primary circuit plateout activity are "steam-induced vaporization" and "washoff'. In both cases, the vehicle for radionuclide release from the primary circuit is water which has entered the primary circuit. In principle, both water vapor and liquid water could partially remove plateout activity (these two removal mechanisms are subsequently referred to as "steam-induced vaporization" and "washoff1, respectively). However, even if a fraction of the plateout activity were removed from the fixed surfaces, there would be environmental release only in the event of venting of helium/steam from the primary circuit. For all but the largest water ingress events the pressure relief valve does not lift.

Moreover, the radiologically important nuclides such as iodine and cesium are expected to remain preferentially in the liquid water which remains inside the primary circuit.

The reactor building/containment structure is the fifth barrier to the release of radionuclides to the environment. Its effectiveness as a release barrier is highly design and event-specific. The US GT-MHR design utilizes a Vented Low-Pressure Confinement (VLPC), and the German modular HTGRs designs utilize similar containment concepts. In contrast, the Japanese HTTR has a high-pressure steel containment building.

The VLPC and similar confinement vented designs may be of limited value during rapid depressurization transients; however, they can be of major importance during core conduction cooldown transients. Under such conditions, the natural removal mechanisms occurring in the reactor building, including condensation, fallout and plateout, are expected to attenuate the release of condensible radionuclides, including radiologically important iodines, by at least an order of magnitude.[37]

[37] International Atomic Energy Agency: Fuel performance and fission product behaviour in gas cooled reactors IAEA TECDOC 978, Vienna, November 1997

*5.1.3 **Releases from TRISO fuel***

• Radionuclide inventories in reactor core are calculated using standard burnup/depletion codes; for example**:**

– ORIGEN for core RN inventories

– GARGOYLE (GA)/VSOP99 (PBMR) for decay heat calculations

RN inventories are specified for:

• Circulating activity in primary coolant

• Plateout activity in primary circuit

• He purification system

Fractional release from the fuel is defined by the formula:

$$(f.r.)_{core} = \frac{C(f.r.)_c + F(f.r.)_F + [1-C-F](f.r.)_D}{AF_{graphite}}$$

$(f.r.)_{core}$ = fractional release from core

C = heavy-metal contamination fraction

$(f.r.)_C$= fractional release from contamination

F = failure fraction:

$(f.r.)_r$= fractional release from failed particles

$(f.r.)_o$= fractional diffusive release from intact particles

AF *graphite* = matrix/graphite attenuation factor

Table 5.1 Dominant Radionuclides in HTGRs

Nuclide	Half Life	Primary Impact
I 131	8 d	Offsite dose, O&M dose
Ag-110m	250 day	O&M dose
Cs-137	30 yr	O&M dose, offsite dose
Cs-134	2.1 yr	O&M dose, offsite dose
Sr-90	28 yr	Offsite dose
Kr & Xe	--	Normal operation gaseous effluent
H-3	12.3 yr	Normal operation liquid effluent; product contamination

*5.1.4 **Significance of Tritium for HTGRs***

According to the experience of AVR:

• Tritium (H-3) will be produced in NGNP by nuclear reactions
  - Neutron activation of impurities (He-3 in coolant; Li in graphite)
  - Ternary Fission (Yield = ~1 x 10-4)
  - Neutron capture in boron control materials

• Some H-3 will accumulate in primary helium
  - Controlled by He Purification System
  - Significant sorption on core graphite

• Fraction of circulating H-3 in He will permeate through IHX &SG with potential to contaminate process gases and steam

• H-3 will contribute to public & occupational exposures
  - Environmental releases from plant (liquid discharge)
  - Contaminated products (e.g., hydrogen, bitumen, etc.)
  - High lithium content in “carbon brick” side reflector (~4 ppm)

• Dominant source of H 3 production

• <50 ppb typical for HTGRs (protium) injection tests

• Displaced H-3 sorbed on core graphite

• Decreased H-3 permeation to secondary H
– Adjusted feed-water chemistry promoted growth of oxide layer on SG tubes reducing H-3 permeation and release
– Large SG leak resulted in large H-3 release from core graphite

• Extensive international data base on HTGR fission product transport available to support HTGR design & licensing, summarized in IAEA TECDOC-978[38].

• Primary data sources

- Previous HTGR R&D programs in USA, FRG, Japan, France, etc.
- Reactor surveillance programs (seven HTGRs constructed)
- On-going R&D programs, especially fuel AGR program

*5.1.5 **Radionuclide Release Barriers- Fuel Kernels***

• Potential release mechanisms
- Fission recoil
- Diffusion
- Hydrolysis (reaction with H2O)

• Controlling parameters
- Fuel temperatures
- Time
- H2O concentration
- Burnup

• Barrier performance
- Fractional gas release function of time/temperature history
- Increased gas release in case of hydrolysis
- Partial diffusive release of volatile fission metals (Ag, Cs > Sr)
- Other radionuclides, including actinides, completely retained

[38] International Atomic Energy Agency: Fuel performance and fission product behaviour in gas cooled reactors IAEA TECDOC 978, Vienna, November 1997

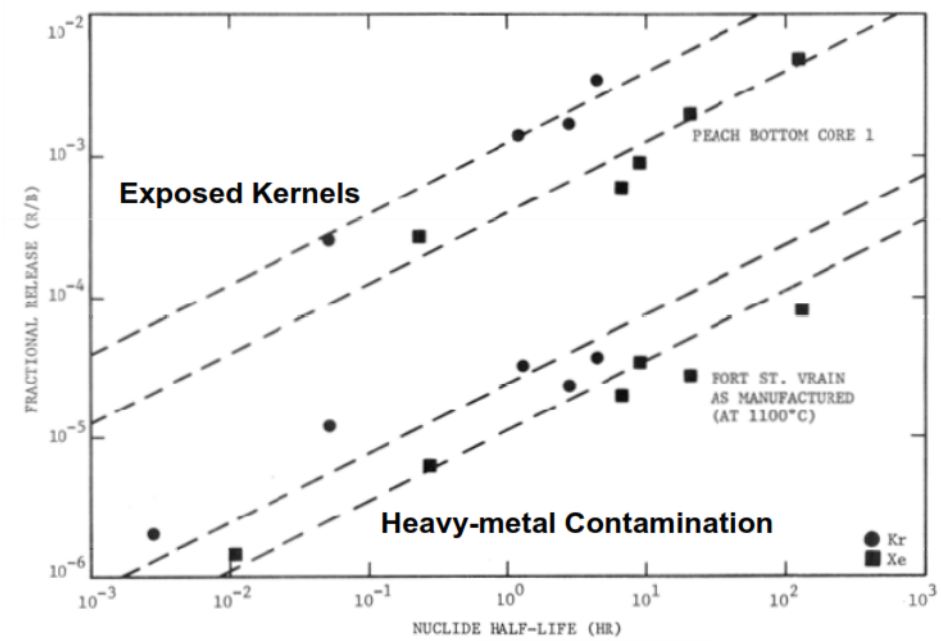


Fig. 5.2 Release rate to birth rate ratio increases with nuclide half-life

Release Rate-to-Birth Rate Ratio generally follows Booth relationship, but deviations havew been observed, especially at lower temperatures and high at lower temperatures and high neutron fluxes (e.g., in HFIR).

5.1.6 Fission Product Release from LEU UO2 Kernels Under Core Conduction Cooldown Conditions

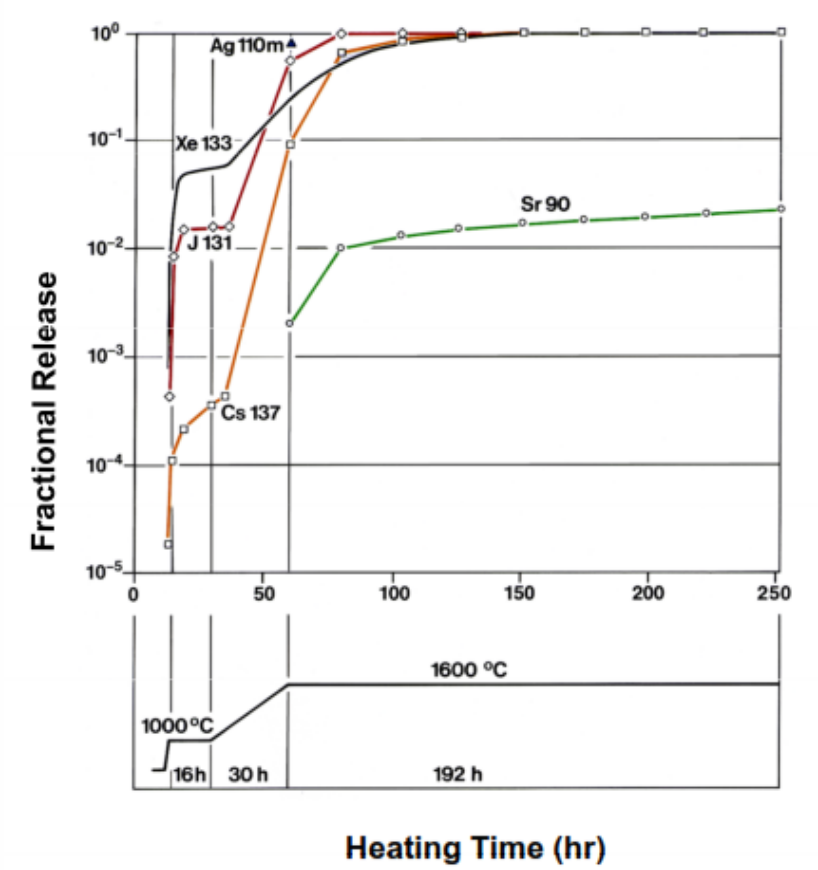


Fig. 5.3 Fission Product Release from LEU UO2 Kernels Under Core Conduction Cooldown Conditions

• Postirradiation heating of FGR LEU UO bare kernels from FRJ2-P28/C6

• Test articles reactivated prior to heating to generate short-lived radionuclides (e.g., I-131)

• FP release behavior as temperature ramped from 1000 to 1600 C:

- Xe-133, I-131 (“J-131”) and Ag-110m rapidly released
- Cs-137 delayed but reaches 100%
- Sr-90 substantially retained for long times

• Kernel release rates expected to increase at higher burnups (low-burnup ThO2 data)

*5.1.7 **Radionuclide Release Barriers- Particle Coatings***

• Potential release mechanisms

– Diffusion through intact coatings

– In-service coating failure

– SiC corrosion by fission products

– SiC thermal decomposition

• Controlling parameters

– Fuel temperatures

– Time

– Fast neutron fluence (Increased FP diffusivities)

• Barrier performance

– Only Ag (and H-3) released by diffusion from intact particles

– No pressure-induced failure of standard particles

– SiC thermochemical failure function of time/temperature

– Gases retained by OPyC with defective/failed SiC

5.1.8 Fission Product Release from LEU UO2 TRISO Particles under Core Conduction Cooldown Conditions

• Postirradiation heating of German LEU UO - TRISO particles in spheres at 1600 & 1800 C

• No complete coating failure (1 particle failure would yield Kr-85 fractional release = ~10-4

• FP release at 1600

Rapid Ag-110m release

Kr-85, Cs-137, and Sr-90 completely retained

• FP release at 1800

Kr-85, Cs-137, and Sr-90 release increasing

Evidence of SiC degradation (expected)

• FP transport in SiC in such tests is ambiguous

Degradation of SiC at T > ~1600 C

FP retention, especially Sr-90, in matrix

• Longer duration tests with intact particles needed to derive effective SiC diffusivities

5.1.9 FP Transport in Coatings

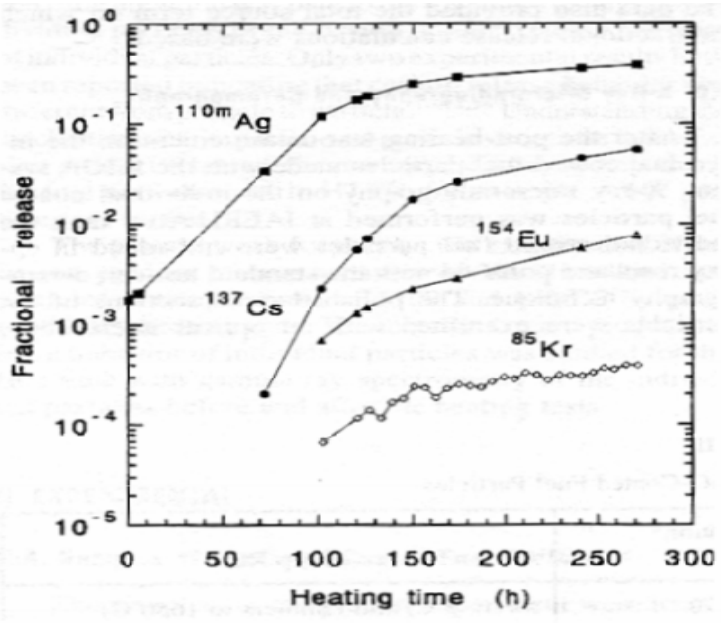
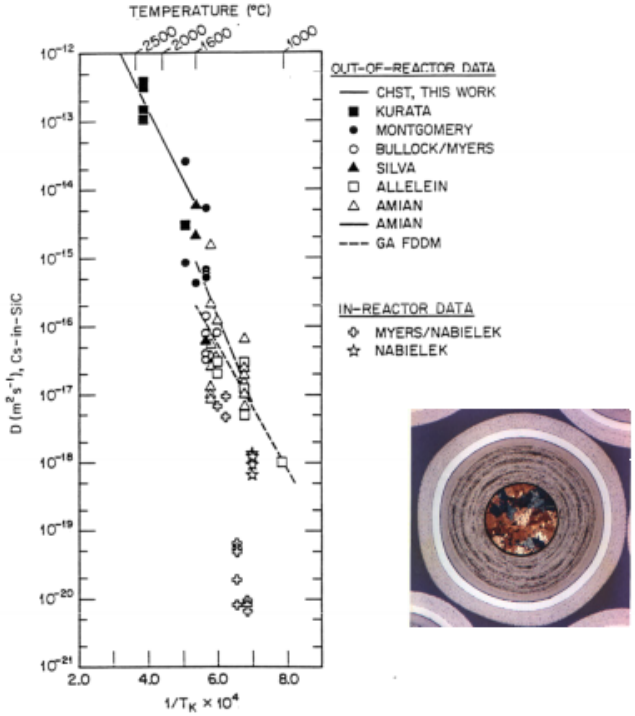


Fig. 5.4 Fractional release of Ag, Cs-137 and Kr-85 from fuel coatings at 1700 oC

In the experiment HRB-22 UO, 25 irradiated fuel particles were recovered from fuel compact after isothermal heating at 1700 oC. Low fractional release of Kr-85 indicates no complete coating failure (1

failed particle = 4% release). Some SiC degradation was observed. Derived fission metal diffusivities in SiC occurred at 1700 °C.



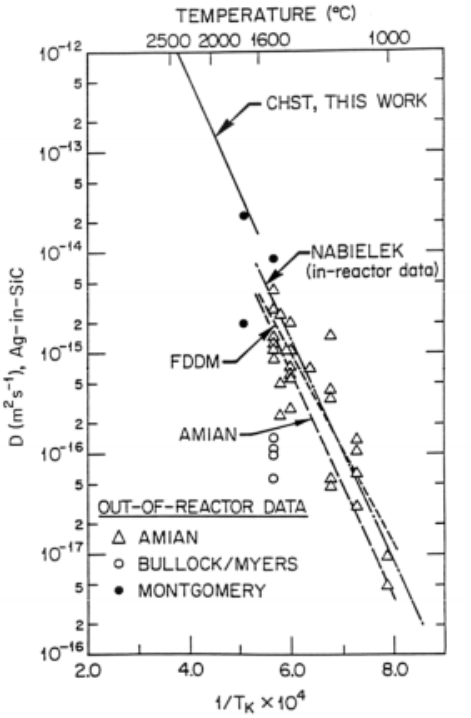


Fig. 5.5 Diffusion coefficients for Cs and AG in SiC

*5.1.10 **Radionuclide Release Barriers Core Matrix/Graphite***

• Potential release mechanisms

– Diffusion/vaporization

– Matrix/graphite oxidation

• Controlling parameters

Temperature – Time

– Fast neutron fluence

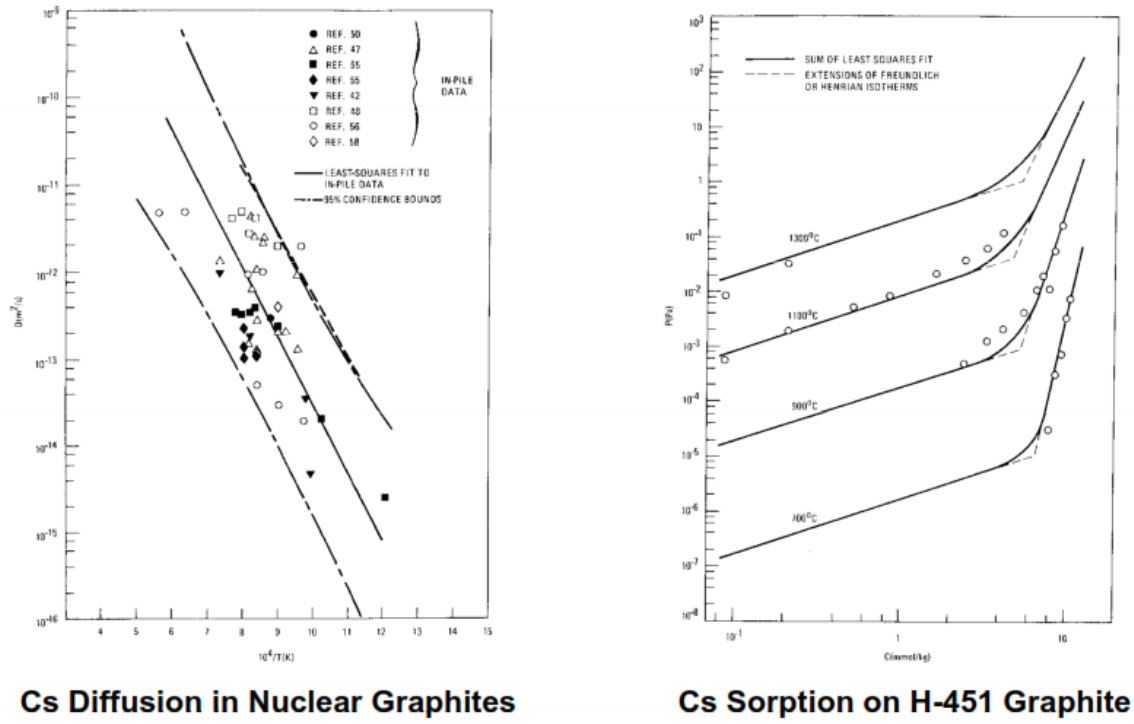
Concentration of H2O

• Barrier performance

Cs and Sr partially released at hotter locations

– Released Cs and Sr partially resorb on cooler graphite

– Sorbed metals assumed to be released by oxidation



Cs Diffusion in Nuclear Graphites **Cs Sorption on H-451 Graphite**

Fig. 5.6 Cs diffusion and sorption in nuclear graphite

*5.1.11 **Radionuclide Release Barriers - Primary Circuit***

• Potential release mechanisms

– Primary coolant leaks

– Liftoff (mechanical reentrainment)

– Primary coolant pressure relief

– Steam-Induced vaporization

– Washoff (removal by liquid H2O)

• Controlling parameters

– Temperatures in primary circuit

– Size/location of coolant leaks

– Particulate matter in primary circuitParticulate matter in primary circuit

– Steam/Liquid H2O ingress and egress

• Barrier performance

– Condensable RNs plate out during normal operation

– Circulating Kr, Xe and H-3 limited by HPS

– Plateout largely retained during rapid blowdowns

– RN holdup due to thermal contraction of gas in vessel

Iodine Sorption on Low-Alloy Steel at 400oC

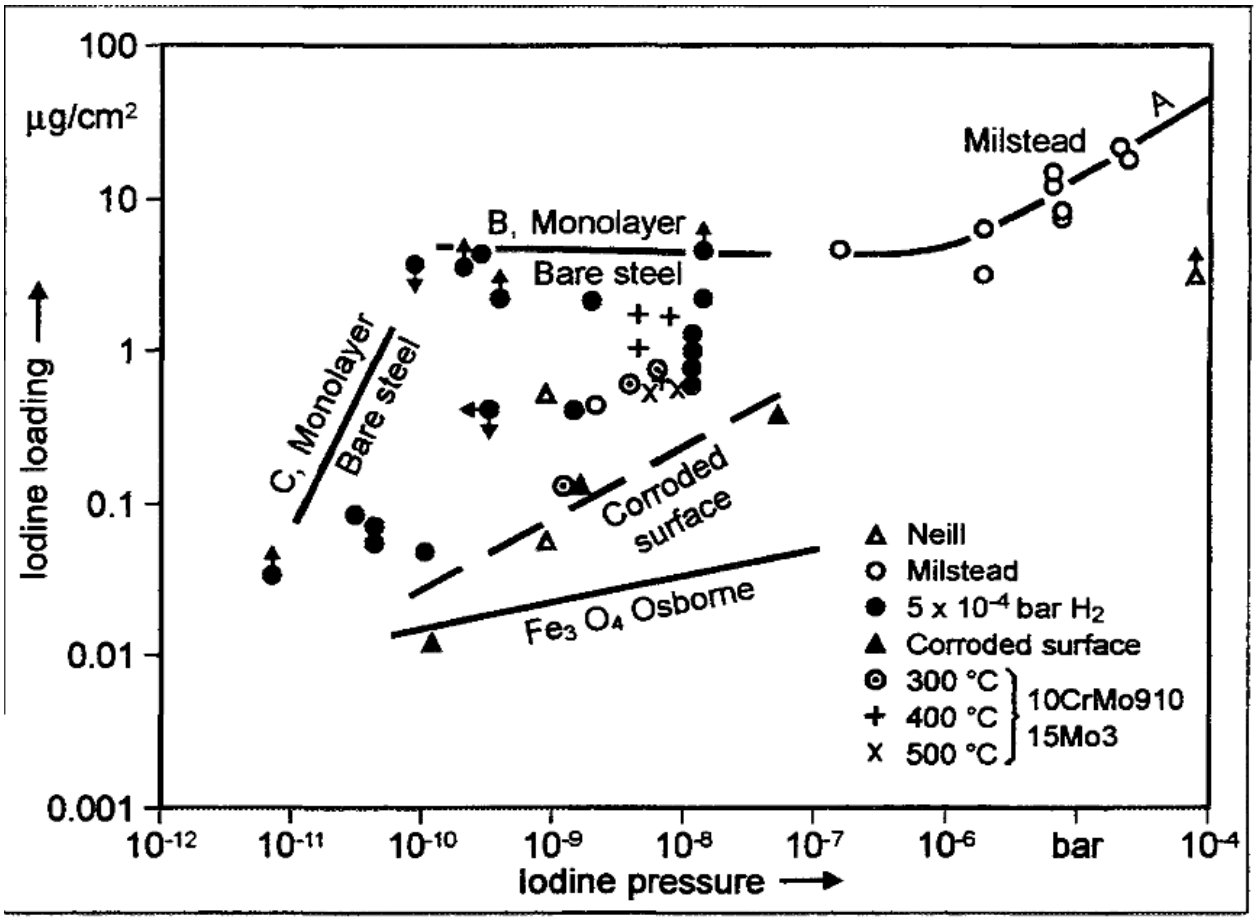


Fig. 5.7: Iodine sorption isotherms, from TECDOC 978[39]

Results of iodine sorption isotherms are summarized in Fig. 5-8. The weak temperature dependence for both 10CrMo910 and 15Mo3 which is surprising for concentrations up to an order of magnitude below a mono layer, seems to confirm the tendency found by Osborne to reach a monolayer.

*5.1.12 **Dust in the primary coolant***

Perhaps the most difficult aspect to understanding and modeling the behavior of condensible fission products in the primary coolant circuit is the potential effects of **aerosols, or "dust",** that may be present therein. Conceptually, it is evident that if the concentrations of circulating and deposited dust in the circuit are very low (e.g., such that the collisions between fission product atoms and circulating particulates are infrequent compared to collisions between fission product atoms and helium atoms and/or fixed surfaces), then dust effects should be unimportant. In the other extreme, if the concentrations of dust are very high, then dust effects should play a dominant role in determining the transport, deposition and reentrainment behavior of condensible fission products.[40]

Wichner, in his review of HTGR plateout and liftoff phenomena, provides a good summary of the available experimental data and the numerous attempts to model aerosol transport effects. In certain cases, including the Peach Bottom HTGR and the AVR, rather high concentrations of dust were

[39] IAEA ibid

[40] IAEA ibid

observed in the primary coolant circuit; however, it is not at all clear that these data are representative of what should be expected for future modular HTGRs, especially those with prismatic fuel elements. Based upon limited FSV data, one might speculate that future HTGRs, especially those with magnetic-bearing circulators or turbocompressors, would be characterized by relatively low levels of dust, once the original construction debris, etc., were removed by prolonged action of helium purification system.

It is also noteworthy here that there is an extensive amount of open-literature data related to aerosol formation, transport, deposition and reentrainment, but it is not apparent that any of it relates directly to the circumstances expected in the primary circuit of an HTGR. In fact, it is not currently possible to state with confidence the nature or concentrations of dust in future HTGR primary circuits which implies large uncertainties in fission product transport behavior will persist until there is significant operating experience with a specific standardized reactor design.

5.1.12.1 British Contribution

There are British data on the transport of metal-oxide aerosols in AGRs, but no data on the effects of such aerosols on fission product transport.

5.1.12.2 FRG Contribution

In a pebble-bed HTGR, graphite dust forms resulting from abrasion during fuel balls recirculation and represents a major transport medium for fission products besides the coolant itself. The **AVR** reactor has been used to conduct several experiments for investigation of atomic or molecular radionuclides in the coolant flow as well as transport and deposition with graphitic or metallic dust.

It can be stated that dust concentrations during stationary, undisturbed operation were in the range of 1 - 2 microg/m3 superposed by concentration peaks that can be higher by some orders of magnitude. The average dust concentration in the AVR over the final operation years was about 5 /microg/m^3.

For the AVR reactor, the dust production rate was estimated to be about 3 kg per operating year with a total dust inventory in the primary circuit of about 60 kg at the end of 1988. Dust concentrations in the coolant can vary over several orders of magnitude with peak values achieved when during transient operation conditions dust is remobilized due to changes in the coolant flow. From such transients, it could be concluded that a fraction of 0.5 - 2 % of the gas-borne dust is deposited per recirculation.

The measurements of Cs-137 activities in graphite dust collected in the THTR-300 from the moisture sensors were taken to assess the fine dust production in 300 days full power operation. The estimated range was 13 - 400 kg with an expected value of 25 kg. The distribution of dust within the primary circuit depending on particle size, coolant velocity, and geometry is, as expected, dominant in the cold gas area. The specific activity is relatively high, 2*10* Bq/g maximum, mainly caused by the radionuclides Co-60, Sr-95, Hf-181, and Pa-235 due to the enhanced particle failure from fuel element fracture. The measurements for the radiologically relevant cesium and silver isotopes were $1.1*10^6$ Bq/g for Cs-137, $5.7* 10^5$ Bq/g for Cs-134, $5.0* 10^5$ Bq/g for Ag-l10 m. The increase of the Cs-137 activity to $4.7*10^6$ half a year later, however, is not that strong as would have been expected from the higher burnup.

According to a prediction made by SIEMENS for the HTR-MODUL, dust will accumulate to 500 - 1000 kg after 32 years of operation. This prediction is based on experience with AVR and on results from experiments with a fuel element feeding system.

The experiments led to an estimate of 10 mg abrasion per transit. The uncertainties with respect to dust effects are remarkably high. In pebble bed HTGRs, most of the dust seems to be produced by friction of fuel element pebbles and mechanical erosion as a result of pebble recirculation, in particular in the fuel handling system, and, therefore, consists mainly of carbon. In addition, some metallic dust has been found containing iron and cobalt and their oxides. Also carburization may play a role in production of some small-sized dust. With respect to dust depletion during normal operation, two different effects must be considered:

• dust transport to surfaces mainly by turbulent diffusion and sticking by adhesion • sedimentation of (probably larger) dust particles in quiescent coolant regions It is a well known fact that adhesively bound particles of diameter < 50 /im are not easy to remove by shear forces. The average size of AVR dust seems to be significantly smaller as indicated by a typical distribution of particle numbers.

One should keep in mind, however, that an agglomeration of individual dust particles on surfaces cannot be excluded especially for high dust concentrations, which could produce particle sizes which are more easily lifted-off by depressurization induced shear forces. Adhesively bound dust can change the plateout behavior of primary circuit surfaces.

The modeling of dust-borne activity can currently not be treated because of major uncertainties concerning production, deposition and interaction with gas-borne or plated out fission products.

Radionuclide activities borne on the dust are significantly dependent on where the dust is located. Assuming half of the dust inventory to be deposited in the fuel cycle system and the other half uniformly distributed between steam generator and cold gas zone, an estimated fraction of 8 % for Cs-137 and of 11 % for Sr-90 out of the totally released activity is bound to dust.

With respect to the dust-borne activity in the AVR, dust from filter experiments was estimated to contain about 10 % of the amount of Cs-137 and Sr-90 released from the core during normal operation.

In the past, the 1-131 content in dust was thought to be nearly negligible. However, recent on-line examination of dust during blower transient experiments has indicated that the AVR dust mobilized during the transient contains concentrations of 1-131 as large as 3.5 GBq/kg. The total amount of iodine in the AVR primary circuit is on the order of 25 GBq.

It is inhomogeneously distributed on the dust in the primary circuit and that, in particular, the dust which contains large concentrations of iodine is preferentially mobilized during transients. A possible reason is that the outer layers of the sedimented dust which come into direct contact with the gas flow contain equilibrium fission product concentrations.

Considering nuclear decay, there will be a concentration gradient toward the bottom dust layers which settled earliest, especially for short-lived nuclides. The outer dust layers seem to be preferentially mobilized. There is no information available about the activity gradient within the settled dust. Therefore, the dust-borne fraction of the radiologically important iodine nuclides cannot be quantified at this time.

It is important to note that the examination of dust specimens removed by wiping from AVR primary circuit surfaces has shown remarkable differences in the specific activities. In particular, dust from fuel element surfaces has a significantly smaller specific activity than dust depleted or adhesively bound on primary circuit components. This was confirmed by the postexamination of irradiated AVR fuel elements, with specific activities on fuel element surfaces found to be one to two orders of magnitude smaller than dust activities. Since dust formation is assumed to proceed mainly by friction of fuel elements, it is not sufficient to consider only the activity on fuel element surfaces as a source of the dust activity.

5.1.13 *Radionuclide Release Barriers -Reactor Building*

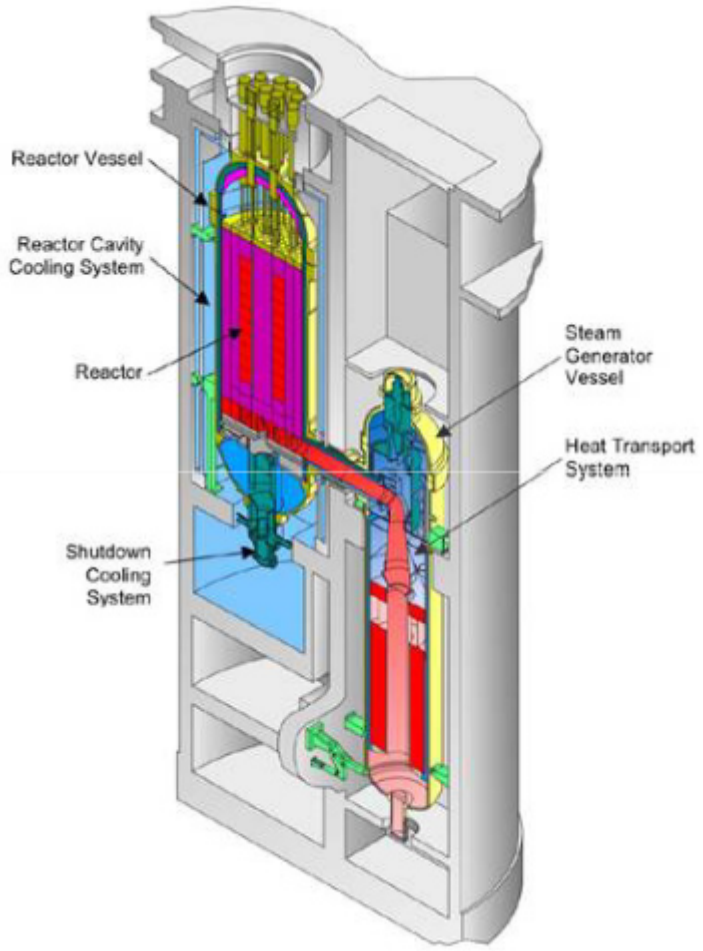


Fig. 5.8 Layout of high temperature reactor with steam generator located side by side with the reactor pressure vessel

• Potential release mechanisms

– Venting through louvers

– Building leakage

• Controlling parameters

– Leak path(s) and rates

– Contaminated particulate matter

– Contaminated steam/liquid H

- Temperatures along leak path(s)Temperatures along leak path(s)

• Barrier performance

– Noble gases decay during holdup

Condensable fission products including I deposit – Condensable fission products, including I, deposit

– Contaminated steam condenses

– Contaminated dust settles out and deposits

5.1.14 *Comparison of FSV Predicted and Measured Fission Metal Release*

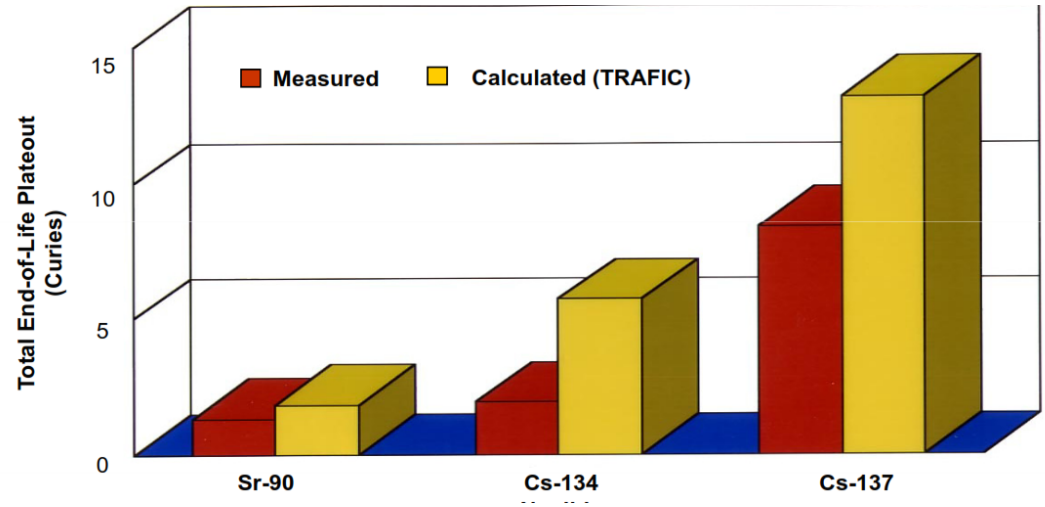


Fig. 5.9 Total end-of-life measured and calculated plateout of Sr-90. Cs-134 and Cs-137

	Sr-90	Cs-134	Cs-137
Measured	1.3	1.9	8.4
Calculated	1.8	5.7	13.2

Predictions of Metallic Release are Accurate and Conservative

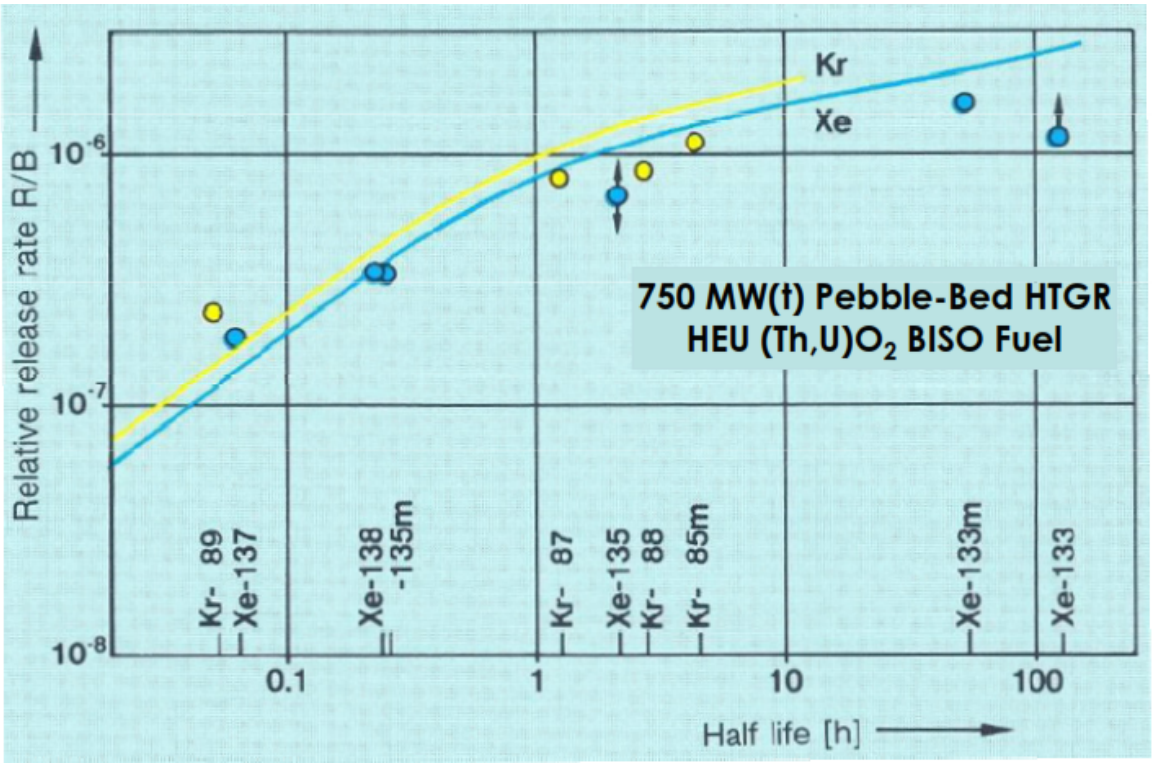


Fig 5.10 Measured and Calculated Noble Gas Release from THTR at 40% Power

Table 5.2 Measured and Predicted Liftoff in COMEDIE BD-1 Loop Test[41]

Nuclide		Cumulative Liftoff Fraction (%)			
		SR = 0.7	SR = 1.7	SR = 2.8	SR = 5.6
I-131	Meas.	0.077	0.10	0.11	0.13
	Pred.	0.15	0.16	0.53	2.1
Cs-137	Meas.	0.014	0.021	0.030	0.11
	Pred.	0.19	0.29	0.48	1.1
Cs-134	Meas.	0.015	0.020	0.028	0.096
	Pred.	0.19	0.29	0.48	1.1
Ag-110m	Meas.	0.015	0.019	0.043	0.23
	Pred.	0.010	0.32	0.90	2.8
Sr-90	Meas.	0.16	0.36	0.56	0.74
	Pred.	0.54	0.56	1.2	4.2

6 Fuel performance and fission product behavior under oxidizing conditions

Among the postulated accidents in the HTGR are those involving water and air ingress. These accidents can lead to extensive degradation of the core components, including the fuel elements, and to a significantly enhanced release of fission products. Water ingress can result from a breach of a steam generator tube and air ingress from breaches in the reactor system that link the internal and external gaseous atmospheres.

Although fuel performance is the central interest, the oxidation of graphite components, other carbonaceous materials and the SiC carbide layer in the fuel particles can play an important role in governing fuel performance. The oxidation of graphite is of primary importance with respect to the structural integrity of the graphite. Much effort has been devoted to the consequences of graphite oxidation.

6.1 General course of the accidents

Given sources of water or air and the mixing of these with the coolant gas, mixtures of water vapor or air with helium circulate in the core passing fuel spheres in the pebble bed core or flowing through coolant channels in the prismatic core. Other graphitic components in the reactor vessel, such as core supports or reflector blocks, are also exposed to the circulating oxidants.

The effect of graphite oxidation depends on the penetration of the oxidant into the graphite as measured from the surface over which the oxidant-containing gases flow. The penetration will depend on the nature of the oxidant, its concentration or partial pressure, the temperature, the type of graphite and ultimately, the time. Decrements in the strength and structural integrity of the graphite core and

[41] HTGR Technology Course for the Nuclear Regulatory Commission May 24 – 27, 2010

support components will depend on the depth of penetration. For a given system and limited times, the penetration depth will decrease with increasing temperature and vice versa for limited times.

In the prismatic core as well as the pebble bed core, the oxidant must penetrate a graphite barrier, having about a 5 mm thickness, to reach the site of the fuel particles. In the prismatic core, the oxidant must penetrate radially outward in the US HTGR and radially inward in the Japanese HTTR. In the pebble bed core, the oxidant must penetrate radially inward through a 5 mm thick, fuel-free coating before encountering the fuel particles in the fuel element (pebble).

When the oxidant reaches the site of the fuel particles, oxidant-fuel interactions are possible along two general paths. Some particles may have failed during the course of the preceding irradiation or fabrication process in a manner that exposed the fuel kernels to the particle exterior. In this case, the oxidant can directly reach the fuel, induce oxidation, and alter the structure of the grains of fuel. The consequence is a significantly enhanced release of fission roducts. A second, more arduous, path involves the oxidation of the outer pyrocarbon coatings of the fuel particles, then oxidation of the more resistant SiC coatings and the inner pyrocarbon and buffer coatings. Frequently the latter two coatings are fragmented; in this case, oxidation of the kernel occurs earlier than otherwise. The second path, more likely in the air ingress accident, could lead to a more serious degradation of the core and to a much larger fission product release than does the first path.

6.2 Experimental and analytical approach to the accidents

6.2.1 ***Experimental Efforts***

Experiments relevant to the HTGR fuel performance during water and air ingress accidents have addressed the response of graphitic materials, coated fuel particles and fuel kernels to oxidants such as O2, CO, and H2O. The specific areas of interest include the oxidation of graphite, the structural properties of oxidized graphite, the oxidation of fuel particle coatings, particularly the SiC coating, and the oxidation of fuel..

6.2.2 ***Analytical Efforts***

The analytical efforts, that is, the development of computer codes based on physical and chemical phenomena and models thereof, have been devoted largely to the area involving the oxidation of graphite and its consequences. The codes and corresponding references applying to graphite oxidation under water and air ingress accident conditions, entitled "Graphite Oxidation Codes", appear in Table 5-1 of IAEA TECDOC 978[42].

6.3 Water ingress accidents

6.3.1 ***Particles with Exposed Fuel***

The effect of water vapor on the release of fission gas from fuel has been investigated by injection of water vapor into carbonaceous systems containing particles with exposed fuel kernels. Such experiments have led to an understanding of the mechanisms of fission gas release in the presence of water vapor and of the dependence of the quantity released and the duration of release on the partial pressure of water vapor, the inventory and the temperature.

The interaction of water vapor with particles having exposed fuel kernels has been investigated both during and after irradiation. Although these types of experiments differ severally, they are potentially complementary. The data accumulated during irradiation indicate two general mechanisms. One reflects a strong dependence of fission gas release on the partial pressure of water vapor below 1 kPa

[42] IAEA ibid

and the other indicates a developing weakening dependence of release on the partial pressure of water vapor above 1 kPa. Such a sequence of dependency has been observed for the fuel UC2.

The general sequential response of the exposed fuel kernels to the injection of water vapor consists of three distinct stages. These three stages are labeled 1, 2, and 3. Stage 0 represents the prehydrolysis reference stage. Stages 1, 2, and 3, respectively, consist of:

(1) a transient release of fission gas with a concomitant increase in the steady-state release,

(2) a period of constant steady-state release, and, upon cessation of water vapor injection,

(3) a decline in the release to prehydrolysis values.

By expanding the time scale of Fig. 6.1, the structure of stage 1 is clarified as shown in Fig. 6.2 for three of the measured isotopes, ie. Kr-87, Kr-88, and Kr-85m. Stages 0 and 2 are represented by mean values of R/B, shown as thick line segments, both within and outside the time segment of Fig. 6.2. These R/B values have a standard deviation of 8 %. In stage 1, the fission gas release increases and then declines after reaching a peak 2 h after starting water vapor injection. The decline in R/B during stage 1 suggests that the gas release in this stage comes from a limited source or availability of fission gas.

The length of stage 1, approximately 9 h in Fig. 6.2, is governed by the slow movement of water vapor to the sites of the exposed kernels in the fuel compact. This is confirmed by the series of 30 sequential I^-H^O reaction spikes appearing on the ionization chamber chart and corresponding to the 30 particles with exposed kernels.

Simultaneously, fission gas is also being released following the UO-IO interaction albeit more slowly and over a long period of time for each kernel. The latter time period is 2h and accounts for the peak in R/B appearing 2 h after initiation of water vapor injection.

During the rise and decline of fission gas release in stage 1, there is an underlying increase in the fractional release between the regions of steady state fission gas release preceding and following stage 1. The time profile of the dashed curve is based on measurements of the time profile of Xe-135; the shape of the underlying curve has been assigned the same functional dependence on time as is observed for Xe-135. The justification for using Xe-135 data is presented below.

During the time interval represented by the data in Fig. 6.2, the fractional release for isotope i, (R/B)i, is given to a good approximation by K/A_{in}. K is a constant reflecting the essential constancy of experimental conditions in the time interval addressed, A_{in} is the decay constant for isotope i and n, a constant, is a measure of the relative fractional concentrations of the isotopes of a given element in a gas sample. A zero value of n corresponds to identical values of the fractional concentrations. As n increases, the fractional concentration of a longer-lived isotope also increases relative to a shorter-lived isotope.

The stages in the response of fuel kernels to the injection of water vapor are distinguished not only by the R/B profiles but also by the profiles of n. This is shown in Fig. 6.3. The mean values of n before and after stage 1 are % 0.35 and n has a peak value of ~ 1.1 in stage 1.

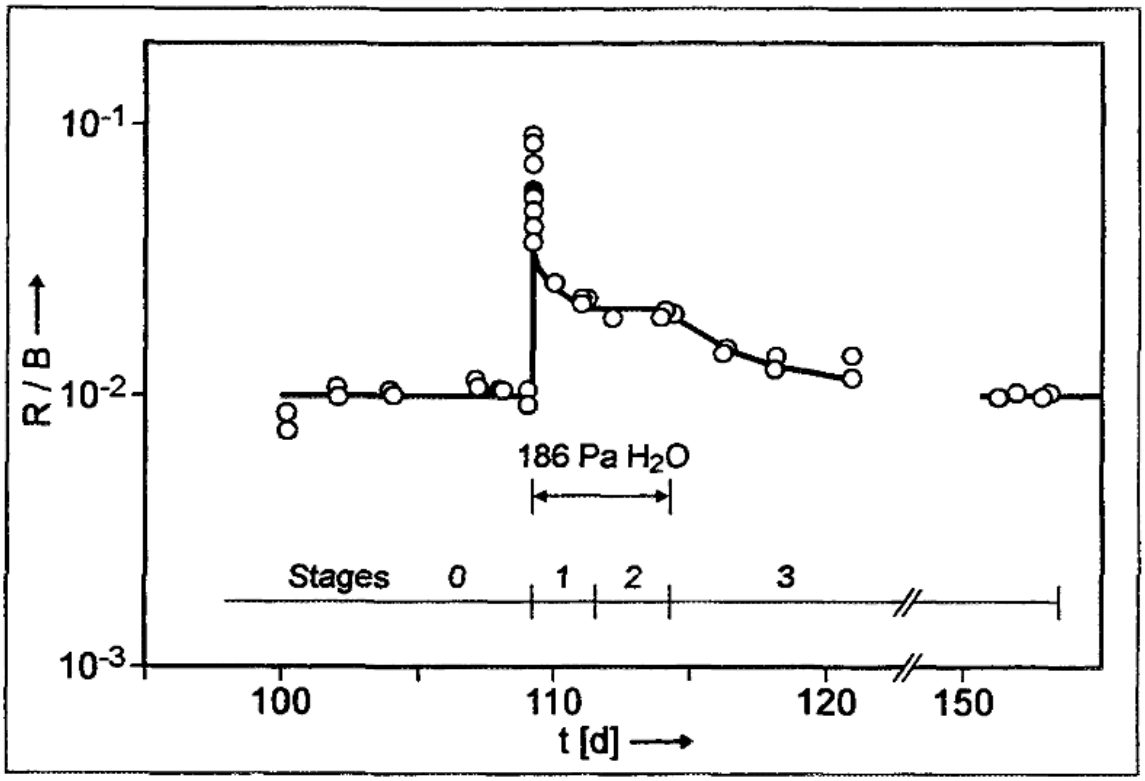


Fig. 6.1: R/B-time profile for Kr-85m before, during, and after a water vapor injection test with 186 Pa of water vapor at 755 °C

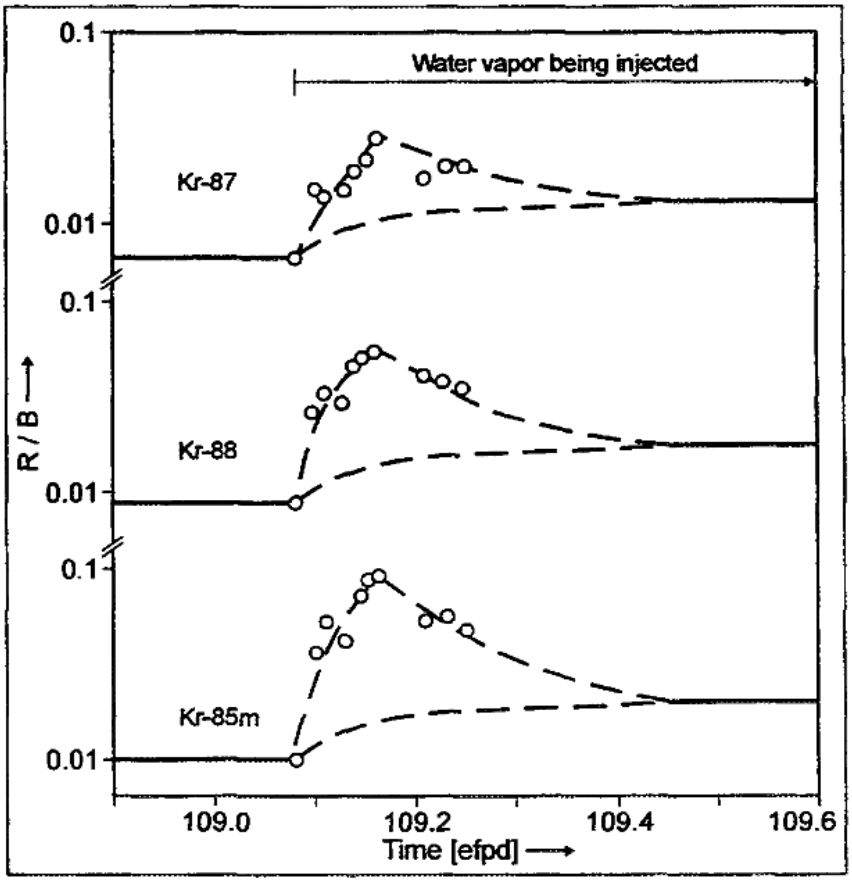


Fig. 6.2: R/B profiles in the early portion of the water vapor injection test at a partial pressure of 186 Pa and at 755 °C in experiment HRB-17

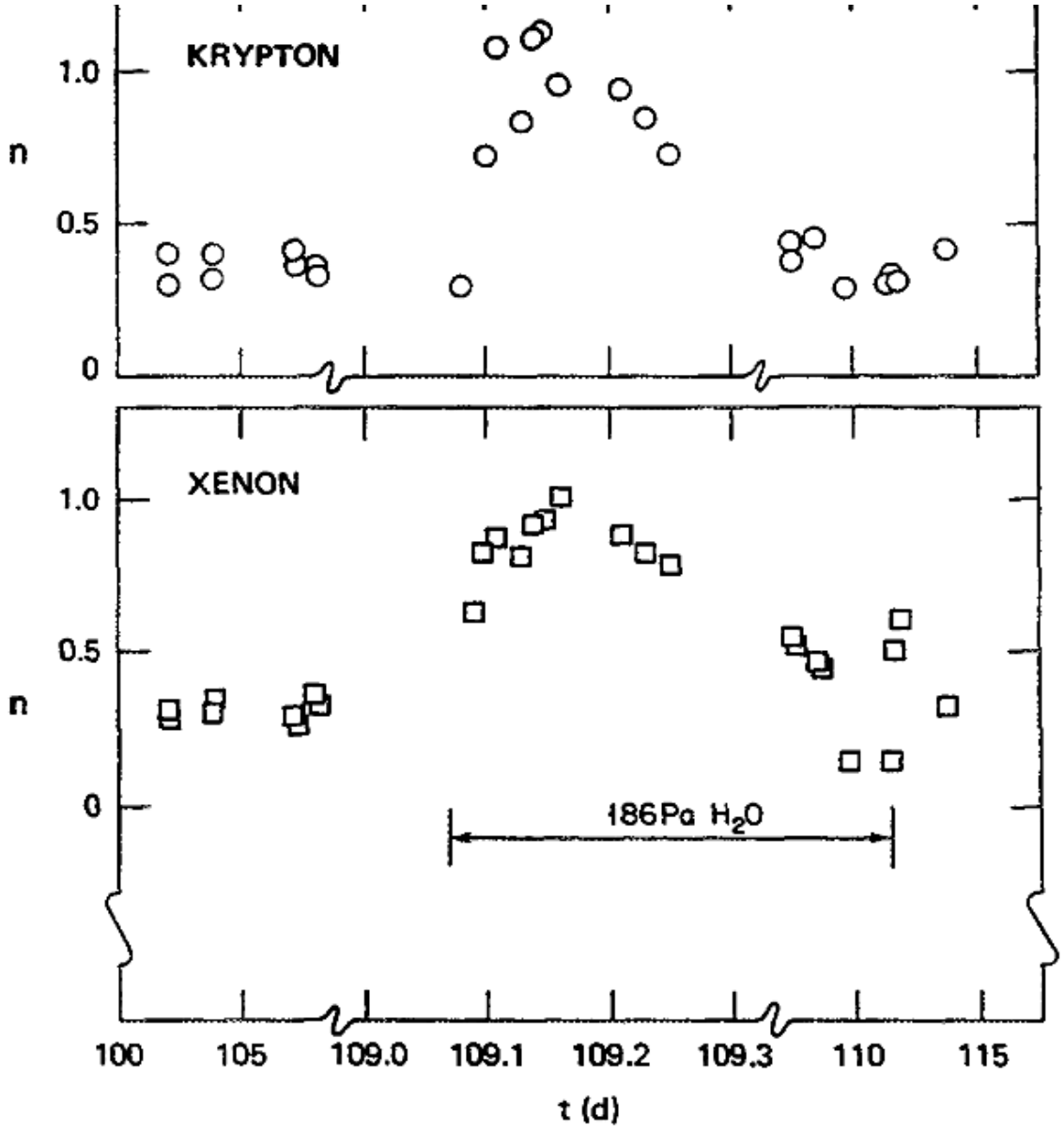


Fig. 6-3: A measure of the relative concentration of isotopes of an element before and during water vapor injection at 755 °C with 186 Pa of water vapor exposed kernels, has n values in the range « 0.2 to « 0.5 and gas collectives have values of n in the range « 1 to % 1.5.

Thus before and after stage 1, the n values of Fig. 6.3 are consistent with diffusive release and during stage 1 are consistent with the predominate release of gas collectives following a breaching of gas bubbles or closed pores.

The curves of Fig. 6.2 can be used to separate the contributions of gas collectives and interstitial fission gas atoms to the quantity of gas released. The Xe-135 release curve on which the underlying curve in stage 1 region is based, represents only diffusive release.

There was no stored Xe-135. Because of the reaction of Xe-135 with neutrons, the effective decay constant was large enough to prevent the accumulation of this isotope in storage during irradiation. Thus the underlying dashed curve in stage 1 represents the continuation of diffusive release in stage 1. The total profile of diffusive release is then established.

The quantity of gas released from collectives, hereinafter referred to as "stored" gas, is determined by summing the differences in R/B values in stage 1 over small increments of time. The data from experiment HFR-BI, over a range of partial pressures of water vapor and temperatures, were used in a

preliminary analysis. The model function previously used was generalized to include a dependence on temperature as well as on the partial pressure of water vapor. This yielded the relation:

$$f = k\, p^{m(T)}\, exp\{-Q/(RT)\}$$

where

f is the fraction of inventory released as stored gas

k is a constant [Pa^{-m}]

m(T) is a temperature dependent exponent, $\mathbf{m(T) = \alpha + \beta T^{-1}}$

α, β are constants

Q is the activation energy for release of stored gas [J/mol]

R is the gas constant, R = 8.314 J/(mol K)

T is the temperature [K]

The above equation is an empirical relation which is meant to describe simply the complicated interplay of the effect of water vapor pressure and temperature on the release of fission gas from exposed fuel kernels. The simplification is centered in the function m(T) which represents the effective declining relative contribution of water vapor to release as the temperature increases. That equation is only valid within the water vapor pressure and temperature ranges of the experimental data.

Table 6.2 Values of h_o as a function of water vapour pressure and element

	Ho = (R/B)i2/(R/B)io			
Partial pressure of water vapour [Pa}	20,6	64,8	186	199
Krypton	2,1	2,0	2,0	2,5
Xenon	2,5	2,5	2,7	3,1

To account qualitatively for the data in Table 6.2, consider a distance of order $(D\,\tau)^{1/2.}$. D is a diffusion coefficient [m2/s] and τ [s] is the lifetime of an isotope (defined as a time by which only a specified small fraction remains radioactively undecayed). The qualitative accounting of the data of Table 6-2 is as follows:

1. If an isotope is within a distance $(D\,\tau)^{1/2}$ of a surface, it can escape from the fuel kernel otherwise it will decay radioactively before escape. When the water vapor interacts with the fuel, the fuel density decreases and D can increase. Consequently the quantity of the isotope escaping will increase by a factor independent of the particular isotope of an element. Furthermore, if increasing the partial pressure of water vapor only increases the depth beneath the surface to which the enhanced D extends, rather than increasing D, then $(D\,\tau)^{1/2}$ remains unchanged as does the quantity of an isotope escaping.

2. When carbide is present and is significantly hydrolyzing, as is the case only in the test with 199 Pa of water vapor, the distortion of the fuel structure is greater than otherwise and D would be expected to increase beyond that observed for an H_2O-UO_2 interaction.

3. For any decrease in the density of the fuel, the effect on diffusivity could be greater for xenon than for krypton atoms because of the larger atomic radius of xenon atoms. The ratio of the atomic volumes

of xenon to krypton, 1.3, is nearly the same as the corresponding average ratio of the values from Table 6.2.[43]

6.3.2 Fission Product Distribution at Initiation of the Accident

In actual safety analyses, the adhesively bound dust-borne activity is assumed to be equally distributed over all surfaces of the primary circuit. Since adhesion forces tend to show only a slight temperature dependence, this assumption is not incorrect as long as the surface structure within the primary circuit is more or less homogeneous. The amount of adhesively bound dust is roughly assumed to be half of the total dust amount. The other half is considered to be sedimented dust. For long-lived nuclides, these values consider a smaller relative dust production rate in the large HTGR compared to the AVR due to a dust filter in the refueling system. Sufficient AVR data for iodine do not exist.

The relative iodine sorption capacity of dust is roughly assumed to be 50 % of that of the long-lived metals. This value considers both the weaker iodine sorption on graphite and its stronger sorption on metallic dust. A reduction of the dust-borne fraction of iodine due to the smaller dust formation rates in modern HTGRs compared to the AVR is not taken into account for short-lived nuclides because then interaction with dust seems to be restricted to the dust layers in contact with the coolant. With respect to the empirical modeling of dust, the uncertainties concerning its production, settlement and interaction with gas-borne or plateout activities are high. In connection with dust contributions to the equilibrium cooling gas activities, it was found that an equilibrium like dust concentration in the coolant is obtained only for uniform flow conditions. Based on experimental evidence from the AVR, a fraction of 10^{-7} of the total mass of dust within the primary circuit is assumed to be gas-bome under equilibrium conditions. For the 200 MW(th) HTR-MODUL, this fraction will contain about $5*10^{-9}$ of the Cs-137 inventory released from core into the primary circuit (or $5*10^{-13}$ of the overall reactor inventory.

Conclusions for the HTGR reactor are as follows:

- less than $2*10^{-4}$ of the inventory is found outside intact coated particles,
- the equilibrium activity within the coolant is extremely low due to low release from the fuel and due to plateout on primary circuit component surfaces,
- the inventories of defective particles and the plateout on the metallic primary circuit surfaces represents most of the activity outside intact coated particles.

6.3.3 Fission Product Transport during Accidents (Reentrainment)

The condensible radionuclides which are plated out in the primary circuit may be partially reentrarned and released to the reactor building during rapid depressurization transients. In the simplest case, volatile fission products that are chemisorbed on primary circuit surfaces could simply desorb; however, since most postulated depressurization transients are characterized by decreasing surface temperatures (i.e., because helium-side heat transfer coefficients are decreasing and feedwater flow is maintained), this removal mechanism is generally perceived to be unimportant. A potentially more significant removal mechanism, especially during rapid depressurizations, is mechanical reentrainment of deposited paniculate matter contaminated by plateout and/or spallation of friable surface films; this mechanical reentrainment is traditionally referred as "liftoff."

Cesium and strontium are strongly bound to cold graphite by chemisorption, whereas a desorption of the small amount of iodine on graphitic surfaces, should not be neglected. With respect to metallic surfaces, a desorption is considered small, because fission product concentrations are small compared to the capacity of the surface for adsorbed fission products.

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[43] IAEA ibid

6.3.4 Desorption of Plateout Activity due to Pressure Drop

Desorption from primary circuit surfaces as one source of activity in a **depressurization accident** is caused by the diminishing partial pressure of fission products in the coolant.

6.3.5 AVR Helium-Cooled Reactor Decontamination Experience

The AVR was a German pebble-bed HTGR which was placed in service in 1967. The primary loop consisted of a gas circulator, the core with fuel spheres and graphite components, and the steam generator. The coolant gas, helium, entered the core at 270 °C and achieved a maximum outlet temperature up to 950 °C. The reactor was operated at a power level of 46 MW(th) and 15 MW(e).

The most reliable data from the AVR decontamination experience were from a circulator, which operated at low temperatures (about 270 °C) prior to the decontamination. In May 1978, water ingressed into the primary loop because of steam generator damage.

After removal of the water, test runs indicated damage to a circulator. The reactor had operated for about 10 years prior to the water ingress. At this time it was decided to remove the circulator for repair. This provided an opportunity to measure plateout behavior of fission products and activation products on the circulator inlet, the circulator wheel, and the wheel cap. These measured activities could then be compared to predictions. In addition, the repair of the circulator necessitated its decontamination to reduce the radiation exposure to personnel. The decontamination effort was broad based with AVR, BBC Baden and HER Wurenlingen (now the Paul Scherrer Institute) involvement, whereby different and complementary methods were investigated. Significant data on radiation levels and decontamination methods were obtained during this effort.

By the time the circulator was removed from the reactor, about 300 days had elapsed after reactor shutdown. All the short lived radioactive isotopes had decayed away. Only long half-life fission products remained and contributed to the measured radiation dose rate from the circulator components. Of the fission products, Sr-90 *(β* decay) represented the greatest specific activity, followed by γ -emitting radionuclides Cs-137 and Cs-134 and Ag-llOm. The *β* activity is a primary concern for skin dose and inhalation doses.

Radiation dose rates were measured using thermo-luminescence dosimeters placed in contact with the surface of the components. In addition, manual measuring devices were used to measure radiation dose rates. They also used wipe samples, scrape samples, microscopic samples using a polisher, and samples using an electrolytic method. The dose rates on contact measured were in the range of 10 to 40 rad/h γ radiation and 100 to 400 rad/h */β* radiation. (Such high radiation dose rates were due to the large fission product plateout as a result of operating a core containing > 50 % BISO fuel at outlet temperatures of 950 °C.) Decontamination treatments were constrained such that only standard media without strong chemical agents could be used to prevent any damage to the mechanical properties of the components which were going to be returned to service. Many decontamination agents were utilized, and the decontamination factors were measured. The wheel cap was fabricated from S 330 A and was not nickel plated. Decontamination of the wheel cap using OX-24 and R-6 solutions provided a decontamination factor of about 75 for cesium and 160 for Ag-llOm. The OX-24 and R-6 solutions are commercially available, proprietary chemical reagent mixtures.

7 Regulations regarding radiological consequences of postulated accidents in MPBR

7.1 NRC recognition that radiation hazards due MPBR are lower than those due to large LWRs

The radiation hazards due to accidents possible in Fort St Vrain were low. The largest collective dose within the distance of 80 km was 0,3 manSv due to rapid depressurization accident, and 0,051 manSv

due to helium purification system regeneration line accident. All other accidents would result in negligible collective doses.

The fact that small LWRs and gas-cooled reactors presented less of a risk and thus could justify smaller EPZs was recognized and reflected in the emergency planning requirements. Specifically, the supplementary information accompanying the rulemaking included NRC's "Position on Planning Basis for Small Light-Water Reactors and Ft. St. Vrain":

"***The Commission has concluded that the operators of small light-water-cooled power reactors (less than 250 MWt) and the Ft. St. Vrain gas-cooled reactor may establish smaller planning zones which will be evaluated on a case-by-case basis.*** This conclusion is based on the lower potential hazard from these facilities (lower radionuclide inventory and longer times to release significant amounts of activity in many scenarios)."

7.2 Protective Action Criteria to be Addressed

As summarized in the NRC' roadmap, SECY 1997-0020, three criteria were used to determine the generic distance for the plume exposure pathway EPZ. These criteria are:

- The EPZ should encompass those areas in which the projected dose from design-basis accidents could exceed the EPA PAGs,
- The EPZ should encompass those areas in which consequences of less severe Class 9 (i.e., core melt) accidents could exceed the EPA PAGs, and
- The EPZ should be of sufficient size to provide for substantial reduction in early severe health effects in the event of the more severe Class 9 accidents.

The EPA *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents* recommends PAGs and corresponding protective actions for both the early and intermediate phases of an atmospheric release of radioactivity. The PAGs are recommended criteria against which projected γ doses to members of the public are compared in determining whether corresponding protective actions should be taken.

From Table 2-1 of the Manual, the PAGs for the early phase of an incident (e.g., hours to days) are:

- Evacuation[44] : 1 – 5 rem[45] and
- Administration of stable iodine: 25 rem
  For the intermediate phase (e.g., days up to about one year beyond the early phase), PAGs for protection of food and water and exposure of the population are provided in Chapter 3 of the Manual.
- Whole body: 0.5 rem and
- Thyroid: 1.5 rem.

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[44] Although evacuation is the preferred action in most cases (within 5 miles of the reactor, page I-51 of NUREG0396), it is recognized that sheltering may be more appropriate in some circumstances considering particular issues such as local weather and mobility concerns (e.g., re-locating nursing home residents). Such actions would normally be initiated when the dose is projected to reach 1 rem, but could be initiated at 5 rem for groups that are less mobile. Moreover, for unusually hazardous environmental conditions, sheltering may be justified for projected doses up to 5 rem for the general population and up to 10 rem for special groups.

[45] This PAG is the sum of the "effective dose equivalent" resulting from exposure to external sources and the "committed effective dose equivalent" incurred from all significant inhalation pathways.. [11] This is the committed dose equivalent to the thyroid from radioiodine

In situations where the only feasible protective actions have high dietary or social costs, Chapter 3 of the Manual describes "emergency" PAGs as:

- Whole body: 5 rem and
- Thyroid: 15 rem.

The PAGs for exposure to deposited radioactivity (Manual Chapter 4, Table 4-1) are:

- Relocation of the general population: 2 rem[46]

and

- Application of simple dose reduction techniques: < 2 rem

From a different source (Table 1 of NUREG-0396), an additional ingestion pathway PAG is stated as:

- Placing dairy cows on stored feed = 1.5 rem to the infant thyroid
- The established threshold for early severe health effects is a 200 rem whole body dose (re: page 8 of SECY 1997-0020 and page I-51 of NUREG-0396) . Per SECY 1997-0020, analyses of LWRs have shown that there is a significant drop in early severe health effects at about a 10-mile distance from the reactor.

In addition, it is recommended that the NGNP project perform the following supplementary analyses

- A PRA evaluation to demonstrate a cumulative mean frequency for sequences resulting in greater than 1 rem total effective dose equivalent (TEDE) over 36 hours at the site boundary, consistent with NRC's review of the MHTGR,
- An evaluation to demonstrate that the plant design is consistent with the prompt accident and quantitative health objective of the 1986 NRC Safety Goal Policy,[47]
- An evaluation to demonstrate that the plant design is consistent with the latent cancer fatality health objective of the Safety Goal Policy.[48]

It is recommended that for the NGNP EPZ reduction program, compliance with all of the above listed criteria (i.e., probabilistic dose analysis and health effect criteria as well as the listed PAGs) be addressed so that the full spectrum of issues (i.e., health effects, food and water protection, meteorology-directional protection planning, etc.) can be addressed when evaluating the feasibility of eliminating or reducing the plume exposure pathway and ingestion pathway EPZs.

7.3 Identification of the Accident Source Term

The foundation for today's emergency planning basis (e.g., the EPA Manual of Protective Action Guides and Protective Actions, NUREG-0396, NUREG-1150) includes an assumed LWR coremelt accident. This is consistent with the "maximum credible accident" requirement for LWR siting and accident analysis 17 . Therefore, LWR PAG dose analyses are a function of a core melt occurrence and the related containment performance (i.e., usually resulting in early releases within 30 minutes to a few hours of accident initiation), site meteorological conditions and the time limit selected for the analysis. The containment analyses include consideration of the source term itself and the timing, magnitude, and characteristics of releases. For HTGRs such as NGNP, the approach to the

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[46] This is the projected sum of effective dose equivalent from external gamma radiation and committed effective dose equivalent from inhalation of re-suspended materials from exposure or intake during the first year

[47] From Section C of the policy statement (51 FR 28044, published August 4, 1986): The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U. S. population are generally exposed.

[48] From Section C of the policy statement (51 FR 28044, published August 4, 1986): The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes. The Policy defines the population considered to be at significant risk as the population living within 10 miles of the plant site.

mechanistic source term and containment function performance needs to be re-evaluated so that the approach to emergency planning can be comprehensively addressed. The NGNP mechanistic source term necessarily will need to include not only releases from the core, but also the expected low probability of a significant release.

7.4 Polish regulations concerning radiation doses at exclusionarea boundary

According to Atomic Law[49] presently inforce in Poland, article 36f states as follows:

Art. 36f. 1. Around the nuclear facility an restricted land-use zone shall be created, according to the principles laid down in the Act of 27 April 2001 – Environmental Protection Law (published in the Journal of Acts of 2008 - Dz. U. No. 25, item 150, with later amendments).

0. The restricted land-use zone around a nuclear facility shall cover an area outside of which:
    1) in operational states of a nuclear facility, including normal operation and anticipated operational occurrences, the yearly effective dose from all pathways of exposure shall not exceed 0.3 milisieverts (mSv);
    2) in case of an accident without core melt the yearly effective dose from all pathways of exposure shall not exceed 10 milisieverts (mSv).

3. When estimating the effective dose, referred to in sec. 2, considered shall be information on:
    1) nuclear facility parameters, including its design and safety measures applied, anticipated releases of radioactive materials to the environment under normal operation conditions, during anticipated operational occurrences and under accident conditions, and type of nuclear materials existing in a nuclear facility;
    2) nuclear facility site, including natural environment conditions in a region of a nuclear facility, in particular: topographical feature, geological structuring, climatic conditions, taking into consideration the most adverse meteorological, hydrological, land and surface slack and flowing water use conditions in a region of a nuclear facility;
    3) nuclear facility operational procedures under normal conditions;
    4) ionising radiation dose distributions for various distances from a nuclear facility, corresponding to operational and accidental states anticipated in the nuclear facility design;
    5) other factors that may influence evaluation of radiation hazards within the restricted land-use zone.

4. Determination of the restricted land-use zone boundaries requires a positive opinion of the President of the Agency.

8 Source terms and radiation doses at exclusion area boundary

8.1 Approach to Source Terms determination

Since the exact configuration of the proposed Pebble Bed HTR for industrial applications in Poland is not defined yet, the analysis of source terms and resulting radiation doses[50] will be based on generic

[49] The Act of 26 November 2000 – Atomic Law (published in the Journal of Acts - Dz. U. of 2007 No. 42, item 276, with later amendments),

data for this type of reactors. A recent report of INL[51] prepared within NGNP program presents the main assumptions and results of such analysis.

8.1.1 Calculation model

This mechanistic approach is used for the following evaluations:

- Demonstration of compliance with 10 CFR 52.47/50.34 **offsite dose** requirements for design basis accidents **(DBA**), with **only safety-related systems, structures, and components** (SSCs) considered.
- Demonstration of compliance with **PAGs for LBEs, with all SSCs** (safety and non-safety) considered.
- Demonstration of compliance with **Polish nuclear safety requirements,** with consideration of differences between Polish and US regulations.

In the absence of complete technical information to support detailed mechanistic source term assessments, the NGNP Project utilized a **simplified model** to perform parametric scoping analyses of the release and retention parameters that define the source terms for a modular HTGR.

The dose is evaluated for each accident as summarized below.

- The distribution of the source term, defined as the radionuclides released from the reactor building to the environment, is determined as a function of the initial core inventory and a release coefficient. The release coefficient is developed by combining the nuclide dependent distributions associated with the release of radionuclides and the attenuation factors associated with the barriers mitigating the release. The source term is calculated based upon fission products released from the fuel during both normal operations and accidents.
- Atmospheric dispersion (χ/Q) and deposition (D/Q) values are determined using the methodology and meteorological information identified in the report, "Engineering Evaluation of χ/Q Values Consistent with Regulatory Guide 1.145."[5] For the purpose of the dose calculation, distances of 400 m, 800 m, 1600 m, 3200 m, and 8000 m are specified.
- The nuclide dependent distribution of the dose conversion factors is then adjusted by applicable χ/Q or D/Q values to derive a distance-weighted dose conversion factor, defined as the dose to an individual at a specified distance per curie of material released. The distance-weighted dose conversion factors are calculated externally to the model in MACCS.
- The results of the probabilistically derived source term and the distance-weighted dose conversion and deposition factors for each nuclide are then combined and summed as shown in the equation below. The results of this are then compared against the NGNP Project selected set of acceptance criteria and against acceptance criteria in force in Poland.

8.1.2 HTGR Configurations, Fission Product Classes and Associated Inventories

The analysis will be done for a single 250 MW(t) 700°C reactor outlet temperature (ROT) pebble bed HTGR module providing high temperature steam via a steam generator.

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[50] Radiation doses will be calculated according to the following simplified formula $Dose_i = Core\ Inventory_i\ (curies) \times Release\ Coefficient_i\ (fraction) \times Dose\ Conversion\ Factor_i\ (rem/curie)$

[51] Scoping Analysis of Source Term and Functional Containment Attenuation Factors February 2012 Idaho National Laboratory, Next Generation Nuclear Plant Project, INL/EXT-11-24034 Revision 1

In the analysis a set of fission product classes will be considered as listed in Table 8.1. Radionuclides within a fission product class are assumed to have the same release and attenuation factors based on physical and chemical properties, with the exception of silver where the two representative isotopes have much different half-lives. The short-lived Ag-111 is distinguished from the much longer-lived Ag-110m in estimating the diffusive release from the fuel under normal operation. Each fission product class contains the key radionuclides, that are expected to account for the majority of the offsite dose.

Table 8.1. Fission product classes.

Fission Product Class	Characteristic Nuclides
Noble Gases	Kr-85, Kr-88, Xe-133
I, Br, Te, Se	I-131, I-133, Te-132
Cs, Rb	Cs-134, Cs-137
Sr, Ba, Eu	Sr-90
Ag, Pd	Ag-110m, Ag-111
Sb	Sb-125
Mo, Ru, Rh, Tc	Ru-103
La, Ce	La-140, Ce-144
Pu, actinides	Pu-239

Inventories for the 250 MW(t) pebble bed are shown in Table 8.2.

Table 8.2. Initial core fission product inventories, in curies.

Fission Product Class	Characteristic Nuclide	Half-Life Days	Decay Constant[a]	Inventory-250MW[b,c]
Noble Gases	Xe-133	5.29	1.52E-06	1.51E+07
	Kr-85	3916.45	2.05E-09	0.79E+05
	Kr-88	0.116667	6.88E-05	0.77E+07
I, Br, Te, Se	I-131	8.04	9.98E-07	0.83E+07
	I-133	0.866667	9.26E-06	1.50E+07
	Te-132	3.25	2.47E-06	1.13E+07
Cs, Rb	Cs-137	10986.5	7.30E-10	0.70E+06
	Cs-134	751.9	1.07E-08	0.79E+06
Sr, Ba, Eu	Sr-90	10585	7.58E-10	0.70E+06
Ag, Pd	Ag-110m	252	3.18E-08	3.23E+04
	Ag-111	7.47	1.07E-06	1.23E+06
Sb	Sb-125	996.45	8.05E-09	0.98E+05
Mo, Ru, Rh, Tc	Ru-103	39.6	2.03E-07	1.50E+07
La, Ce groups	Ce-144	103660	7.74E-11	0.97E+07

	La-140	1.675	4.79E-06	1.36E+07
Pu, actinides	Pu-239	8800150	9.12E-13	1.94E+03

Decay constant (λ) from http://users.ictp.it/~pub_off/lectures/lns020/Nichols/Nichols.pdf

Inventory based on 91117 Rev 0 *NGNP Contamination Control Study*, General Atomics, 2008

Plutonium inventory from General Atomics (350 MW).

8.1.3 Tristructural-isotropic (TRISO) Fuel Fabrication Quality

The level of defects in tristructural isotropic (TRISO) fuel (or the fuel "quality") resulting from the fuel fabrication process is important in establishing the overall source term for an HTGR. Specifications are placed on fuel fabrication to limit the levels of heavy metal contamination (HMC) and SiC coating defects so that overall radionuclide control requirements are met. The values selected for heavy metal contamination (within the matrix that is not coated) and SiC defects (in the multiple coatings of each particle) reflect current fuel fabrication experience in the United States.

The heavy metal contamination fractions are the same as those used historically by U.S. reactor designers in their design assessments. The SiC coating defect fractions are about 5 times lower than those historically required by the reactor designer. A lower SiC coating defect level results in a lower metallic fission product (e.g., cesium and strontium) release into the graphite and fuel matrix during normal operation.

The fuel defect fraction estimates are presented in Table 8.3 and were assumed to be normally distributed.

Table 8.3. Fuel defect fractions for all reactor configurations.

Fission Product Class	Fabrication			
	Fraction Heavy Metal Contamination		Fraction SiC Coating Defects	
Confidence Limit	50%	95%	50%	95%
Noble Gases	1E-5	2E-5	NA	NA
I, Br, Se, Te	1E-5	2E-5	NA	NA
Cs, Rb	1E-5	2E-5	1E-5	3E-5
Sr, Ba, Eu	1E-5	2E-5	1E-5	3E-5
Ag, Pd	1E-5	2E-5	1E-5	3E-5
Sb,	1E-5	2E-5	1E-5	3E-5
Mo, Ru ,Rh, Tc	1E-5	2E-5	1E-5	3E-5
La, Ce	1E-5	2E-5	1E-5	3E-5
Pu, actinides	1E-5	2E-5	1E-5	3E-5

I in HMC = 8.3E6 Ci x 2E-5 = 1.66 E2 Ci

8.1.4 Fission Product Release from Fuel/Graphite during Normal Operation

The normal operations source terms for modular HTGRs are developed from four major sources:

1) heavy metal contamination in the fuel,

(2) particles with SiC coating defects (which release fission metals but not fission gases and halogens),

(3) incremental in-service fuel failures that occur under normal operation; and

(4) diffusional release through intact coatings.

Fuel failure is defined as a loss of the specified retention capability of all the coating layers. However, significant retention of radionuclides is maintained with the kernel, which is accounted for in the analysis. Reactor designers calculate fuel performance accounting for some incremental level of particle coating degradation and failures. The fuel qualification program will provide actual data on a large statistically significant population of TRISO fuel particles to better quantify this value.

Even for similar reactor sizes, HTGR source terms vary depending upon reactor type, thermal power rating, and outlet temperature. For this study, the evaluation assumed the pebble bed designs had a peak normal operating fuel temperature that is 100 to 150°C lower than that for a prismatic reactor at the same power rating. Additionally, the higher outlet temperature designs have higher fuel temperatures and higher average core temperatures, which results in a higher normal operational source term, a greater amount of potential incremental fuel failure during normal operation, and a larger inventory of fission products captured in the graphite and circulating in the primary coolant. These effects were accounted for in the analysis.

To account for uncertainty in fuel and barrier performance, the review team estimated 50% and 95% confidence values for the incremental in-service failure fractions and attenuation factors upon which the mechanistic source terms are based. Incremental in-service failure fractions were assumed to be normally distributed. Attenuation factors were assumed to be lognormally distributed.

Because of the robust nature of the TRISO fuel, demonstrated through historical German and recent U.S. AGR Program experience, minimal particle failures are expected under anticipated normal NGNP design service conditions (temperature, burnup, fast fluence). Thus, the particle failure fractions under normal conditions represent statistical lower limits based on testing of fuel as part of the fuel development and qualification program. Because a large amount of fuel will be tested under prototypic conditions, the statistically based failure rate is 7^{-10} times less than the historical reactor design requirements, thus reducing the source term associated with incremental particle failures under normal conditions.

*8.1.5 **Attenuation Factors during Normal Operations***

Attenuation factors (AF) represent the retention capability of each barrier to fission product release. These AFs were estimated based on HTGR fuel testing results and radionuclide transport predictions developed for previous HTGR designs such as the modular high temperature gascooled reactor (MHTGR) and Pebble Bed Modular Reactor (PBMR). The results for pebble bed HTR are presented in Table 8.5. For the noble gases, AFs associated with the matrix and the helium pressure boundary (HPB) were set to 1 (no retention). Of the fission products released to the coolant with the helium pressure boundary, noble gases were assumed to remain in circulation (AF = 1), while all others were assumed to plate out in the HPB during normal operation.

For the pebble bed HTGR cases, the lower normal operating fuel temperature results in higher attenuation factors associated with heavy metal contamination, the kernel, and diffusive releases. However, the pebble bed does not have fuel element graphite, so the total fuel element AFs are lower than in prismatic fuel reactors, reflecting the lower fission product retentiveness of the fuel matrix of the pebble spheres compared to that of the combined prismatic fuel matrix of the compacts and the fuel element graphite.

In general, higher outlet temperatures tend to increase projected particle failure and decrease the in-core retention of the fission products. Thus, a larger inventory of fission products is released from the core and helium pressure boundary, and the reactor building plays a more important role in mitigating the release to the environment.

The pebble bed design, with a lower power rating and a lower outlet temperature, has correspondingly lower inventories, despite having a lower graphite attenuation factor than the prismatic reactor.

Table 8.4. Mean values for I-131, Cs-137, and Sr-90 inventories (curies) released to the helium pressure boundary and retained in the fuel matrix and graphite.

Reactor Design Configuration	I-131, Curies		Cs-137, Curies		Sr-90, Curies	
	In Fuel Matrix and Graphite	In Helium Pressure Boundary	In Fuel Matrix and Graphite	In Helium Pressure Boundary	In Fuel Matrix and Graphite	In Helium Pressure Boundary
250 MW(t) Pebble Bed 700°C ROT	-	9	34	3	1250	0.4

Table 8.5. 250 MW(t) pebble bed 700°C ROT—normal operations—release fractions and attenuation factors.

Nuclide	Sources of Fission Product Release						Barriers to Fission Product Release (Attenuation Factors)							
	Fract. HM Contamination		Fraction SiC Defects		In-Service Failure		Heavy Metal Contamination		Kernel		Diffusive Release thru coating		Graphite	
	50%	95%	50%	95%	50%	95%	50%	95%	50%	95%	50%	95%	50%	95%
Xe-133	1E-05	2E-05	NA	NA	7E-06	2E-05	15	4.5	75	25.5	1E+06	1E+05	1	1
Kr-85	1E-05	2E-05	NA	NA	7E-06	2E-05	15	4.5	75	25.5	1E+06	1E+05	1	1
Kr-88	1E-05	2E-05	NA	NA	7E-06	2E-05	15	4.5	75	25.5	1E+06	1E+05	1	1
I-131	1E-05	2E-05	NA	NA	7E-06	2E-05	15	4.5	75	25.5	1E+06	1E+05	1	1
I-133	1E-05	2E-05	NA	NA	7E-06	2E-05	15	4.5	75	25.5	1E+06	1E+05	1	1
Te-132	1E-05	2E-05	NA	NA	7E-06	2E-05	15	4.5	75	25.5	1E+06	1E+05	1	1
Cs-137	1E-05	2E-05	1E-05	3E-05	7E-06	2E-05	1	1	20	8	1E+06	1E+04	5	1
Cs-134	1E-05	2E-05	1E-05	3E-05	7E-06	2E-05	1	1	20	8	1E+06	1E+04	5	1
Sr-90	1E-05	2E-05	1E-05	3E-05	7E-06	2E-05	1	1	200	80	800	200	50	15
Ag-110m	1E-05	2E-05	1E-05	3E-05	7E-06	2E-05	1	1	8	4	40	4	1	1
Ag-111	1E-05	2E-05	1E-05	3E-05	7E-06	2E-05	1	1	8	4	40	4	1	1
Sb-125	1E-05	2E-05	1E-05	3E-05	7E-06	2E-05	1	1	8	4	1E+06	1E+04	10	1
Ru-103	1E-05	2E-05	1E-05	3E-05	7E-06	2E-05	1	1	2000	120	1E+06	1E+05	100	30
Ce-144	1E-05	2E-05	1E-05	3E-05	7E-06	2E-05	1	1	2000	120	1E+06	1E+05	100	30
La-140	1E-05	2E-05	1E-05	3E-05	7E-06	2E-05	1	1	2000	120	1E+06	1E+05	100	30
Pu-239	1E-05	2E-05	1E-05	3E-05	7E-06	2E-05	1	1	4000	400	1E+06	1E+05	1000	100

*8.1.6 **Accidents Selected for Evaluation***

Based on safety analyses performed for prior gas-cooled reactor studies, breaks in the HPB and water ingress events pose the greatest challenges with respect to offsite dose consequences. As discussed further in Appendix A, previous analyses of these events were reviewed to aid in developing the scenarios to be considered in this model for evaluating the mechanistic source terms. The results from these past assessments also provided additional data for the expert panel to estimate barrier performance and against which the panel's intermediate results were benchmarked. The two scenarios developed and key conditions are summarized here:

- ***Break in HPB with loss of forced cooling***
  - Leak or break in the HPB piping up to the largest connecting pipe
  - Reactor trip
  - Loss of heat transport to the energy conversion
  - Loss of shutdown cooling
  - Immediate depressurization of helium in the helium pressure boundary
  - Opening of the reactor building vent to relieve helium pressure

This event included an early release of the circulating activity with liftoff of plated out material from the HPB due to the depressurization followed by a subsequent long term release because of the core heatup from the fuel and graphite and then from the helium pressure boundary.

- ***Water ingress event (for design configurations with a steam generator)***
  - Steam generator tube break
  - Reactor trip
  - Loss of heat transport to the energy conversion
  - Loss of shutdown cooling
  - Detection of water ingress
  - Isolation of the steam generator main steam and feedwater lines
  - Over-pressurization of the HPB through the vessel system relief valve
  - Opening of the reactor building vent to relieve helium and water/steam pressure.

This event included an early release of the circulating activity with washoff of plated out material from the HPB due to the over-pressurization followed by a subsequent long term release due to the core heatup from the fuel and graphite and then from the HPB.

The barrier performance characteristics required as input in the model simulations for these scenarios were then established to be representative of the significant accidents based on knowledge of the requisite phenomenology and results from similar events previously analyzed in HTGR safety assessments.

A third key scenario that has been determined to be risk-significant in prior analyses of the MHTGR and the PBMR, was a ***small HPB leak followed by a depressurized conduction cooldown***. In this event, the HPB depressurization extends for up to several hundred hours, which provides a transport mechanism for the delayed radionuclide release from the fuel as a result of the core heatup but this event does not include the release of plateout material within the helium pressure boundary. This event was bounded in the accidents analyzed by including the initial release of plateout material due to lift off and the release of radionuclides due to core heatup following the depressurization phase (even in the

absence of a strong driving mechanism for the release). The analysis further assumed shorter release durations, which resulted in higher atmospheric dispersion (X/Q) values than would be expected for a release lasting many hours. Thus, for the very unlikely event of a small leak without mitigative actions taking place, such as restart of forced cooling and/or intentional helium depressurization through the helium purification system to helium storage, the event would be adequately addressed by the analyzed break scenario and was not selected for further study as part of this initial evaluation.

The configurations for pebble bed HTGR and the selected accident scenarios simulation are presented in Table 8.6.

Table 8.6 Mechanistic source term events for model simulations

Reactor Configuration	Accident Scenario
250 MW(t) Pebble Bed, 700°C ROT	Break in Helium Pressure Boundary
250 MW(t) Pebble Bed, 700°C ROT	Water Ingress Event

8.2 Results of source terms evaluation for pebble bed HTR

8.2.1 Releases and Attenuation Factors Under Accident Conditions

For modeling the accident conditions the mechanistic source term is assumed to comprise two distinct phases: a short-term (prompt) release during depressurization of the fission products from normal operations, and a long-term (delayed) release primarily due to heatup of the fuel.

For the prompt release, the entire inventory of circulating activity (noble gases) is assumed to be released, and a portion of the material plated out within the HPB is assumed to be lifted off (in the event of a dry break in the helium pressure boundary) or washed off (in a water ingress event). Table 8.7 lists the short-term 50 and 95% confidence AFs estimated for the HPB and the reactor building for a 250 MW(t) pebble bed reactor with a 700°C ROT under a break in the HPB accident scenario.

The AF estimated for the early depressurization phase in the break in the helium boundary scenario is based on the shear force ratio on surfaces of the HPB and is based on results from the COMEDIE in-pile loop tests.[9] The COMEDIE tests indicated an AF of 1000 at a shear ratio of 1.7. Since this is a shear ratio higher than calculated for HPB breaks in modular HTGR designs, the 50% value of 200 used here is judged to be acceptable in light of the lack of detailed plant configuration and design.

For the long-term release, the core is assumed to heat up for 50 hours, after which time a release of nuclides is assumed for the next 40 hours.

Under accident conditions, intact fuel particles do not release fission products except silver. Releases from the core under accidents are dominated by release from heavy metal contamination, in-service incremental failures, and release of fission products retained in the graphite under normal operation.

Under the accident scenario of a break in the helium pressure boundary, the AF of the HPB for a long-term heat up was reduced relative to the AF under prompt release conditions. The long-term accident AFs take into account the very low flows expected at this phase of the event and the eventual contraction of the helium in the reactor vessel as the core cools down and the flow reverses into the helium pressure boundary. Preliminary calculations performed by PBMR for a medium break scenario suggest that AFs greater than 750 are achievable due to the lack of pressure driving force for radionuclide transport from the core through the HPB into the reactor building. In the cases where there is a pressure driving force (such as a very small break) the AF was only 1.7.[10] Thus, the AFs of 5^{-10} were also judged to be

acceptable estimates given the limited knowledge associated with the accident phenomenology and design detail available.

In the break in the HPB scenarios, the reactor building AF of 10 was judged to be acceptable in light of the lack of building design and the desire to focus the reactor building requirements on structural protection of the reactor as opposed to radionuclide retention. The PBMR calculations indicate long term AFs in excess of 2000 for a medium size break and 20 for a very small break may be achieved. In these calculations, the reactor building was assumed to leak at 100% per day at 0.1 bar overpressure. As such, the AFs are not associated with any specific fission product retention mechanisms or leak tightness of the reactor building but rather with the lack of any strong flow to transport the gas through the reactor building to the environment.

In the case of the water ingress event, the uncertainties are larger because of the lack of data. However, because the water is expected to remove a large fraction of the plateout inventory, the AFs for the HPB were set much lower than in the HPB break case. Because condensation of the moisture in the reactor building during transport is expected to mitigate the escape of fission products from the building, a higher AF was estimated for the reactor building. As discussed above, in general, the AFs for the reactor building (RB) were selected to allow flexibility in the final design details to accomplish the primary function of the building to provide structural protection of the reactor to maintain core geometry for control of heat generation and core heat removal to the reactivity cavity cooling system (RCCS). Thus, the AFs are not overly aggressive for the RB's supportive function of radionuclide retention.

8.2.1.1 Accident Scenario: Break in Helium Pressure Boundary

Table 8.7. 250 MW(t) pebble bed 700°C ROT—break in HPB—short term attenuation factors.

Nuclide	Short Term Barriers to Fission Product Release (Attenuation Factors)			
	Accident – HPB		Accident – Reactor Building	
	DF 50%	DF 95%	DF 50%	DF 95%
Xe-133	1	1	1	1
Kr-85	1	1	1	1
Kr-88	1	1	1	1
I-131	200	20	2	1
I-133	200	20	2	1
Te-132	200	20	2	1
Cs-137	200	20	2	1
Cs-134	200	20	2	1

Sr-90	200	20	2	1
Ag-110m	200	20	2	1
Ag-111	200	20	2	1
Sb-125	200	20	2	1
Ru-103	200	20	2	1
Ce-144	200	20	2	1
La-140	200	20	2	1
Pu-239	200	20	2	1

Table 8.8. 250 MW(t) pebble bed 700°C ROT— break in HPB—long term release fractions and attenuation factors.

Nuclide	Sources of Fission Product Release						Barriers to Fission Product Release (Attenuation Factors											
	Fract. HM Contamination		Non-Intact (In-service failure)		Incremental Failure – Accidents		HM Accident Release		Non-intact Accident Release		Intact Accident Release		Accident – Graphite		Accident – HPB		Accident – Reactor Building	
	50%	95%	50%	95%	50%	95%	DF 50 %	DF 95 %	DF 50%	DF 95%	DF 50%	DF 95%	DF 50%	DF 95%	DF 50%	DF 95 %	DF 50%	DF 95 %
Xe-133	1E-05	2E-05	7E-06	2E-05	3E-05	8E-05	2	1	10	5	1E+07	2E+06	1	1	5	2	1	1
Kr-85	1E-05	2E-05	7E-06	2E-05	3E-05	8E-05	2	1	10	5	1E+07	2E+06	1	1	5	2	1	1
Kr-88	1E-05	2E-05	7E-06	2E-05	3E-05	8E-05	2	1	10	5	1E+07	2E+06	1	1	5	2	1	1
I-131	1E-05	2E-05	7E-06	2E-05	3E-05	8E-05	2	1	10	5	1E+07	2E+06	1	1	5	2	10	2
I-133	1E-05	2E-05	7E-06	2E-05	3E-05	8E-05	2	1	10	5	1E+07	2E+06	1	1	5	2	10	2
Te-132	1E-05	2E-05	7E-06	2E-05	3E-05	8E-05	2	1	10	5	1E+07	2E+06	1	1	5	2	10	2
Cs-137	1E-05	2E-05	1.7E-05	5E-05	3E-05	8E-05	1	1	1	1	1E+07	2E+06	10	3	10	3	10	2
Cs-134	1E-05	2E-05	1.7E-05	5E-05	3E-05	8E-05	1	1	1	1	1E+07	2E+06	10	3	10	3	10	2
Sr-90	1E-05	2E-05	1.7E-05	5E-05	3E-05	8E-05	1	1	1	1	1E+06	2E+05	100	30	10	3	10	2
Ag-110m	1E-05	2E-05	1.7E-05	5E-	3E-05	8E-05	1	1	1	1	5E+02	1E+02	1	1	10	3	10	2

				05														
Ag-111	1E-05	2E-05	1.7E-05	5E-05	3E-05	8E-05	1	1	1	1	5E+02	1E+02	1	1	10	3	10	2
Sb-125	1E-05	2E-05	1.7E-05	5E-05	3E-05	8E-05	1	1	1	1	1E+06	2E+05	100	30	10	3	10	2
Ru-103	1E-05	2E-05	1.7E-05	5E-05	3E-05	8E-05	1	1	100	50	1E+07	2E+06	10	3	10	3	10	2
Ce-144	1E-05	2E-05	1.7E-05	5E-05	3E-05	8E-05	1	1	100	50	1E+07	2E+06	100	30	10	3	10	2
La-140	1E-05	2E-05	1.7E-05	5E-05	3E-05	8E-05	1	1	100	50	1E+07	2E+06	100	30	10	3	10	2
Pu-239	1E-05	2E-05	1.7E-05	5E-05	3E-05	8E-05	1	1	1000	500	1E+07	2E+06	1000	300	10	3	10	2

Table 8.9. 250 MW(t) pebble bed 700°C ROT—break in HPB—source term (curies).

Nuclide	Release to Environment Short Term				Release to Environment Long Term			
	DBA	50%	Mean	95%	DBA	50%	Mean	95%
Xe-133	1.17E+01	1.17E+01	1.17E+01	1.17E+01	7.07E+01	2.06E+01	2.70E+01	7.07E+01
Kr-85	6.07E-02	6.07E-02	6.07E-02	6.07E-02	4.80E-01	1.40E-01	1.84E-01	4.80E-01
Kr-88	6.10E+00	6.10E+00	6.10E+00	6.10E+00	1.98E-04	5.74E-05	7.53E-05	1.98E-04
I-131	3.51E-01	1.68E-02	4.95E-02	1.88E-01	4.37E+01	1.22E+00	2.59E+00	9.28E+00
I-133	5.59E-01	2.80E-02	8.21E-02	3.12E-01	1.78E+01	4.99E-01	1.04E+00	3.69E+00
Te-132	4.61E-01	2.15E-02	6.54E-02	2.47E-01	4.44E+01	1.27E+00	2.65E+00	9.48E+00
Cs-137	7.97E-01	4.15E-02	1.18E-01	4.40E-01	2.98E+00	4.59E-02	1.36E-01	5.26E-01
Cs-134	1.22E-01	6.39E-03	1.79E-02	6.63E-02	3.28E+00	5.04E-02	1.53E-01	5.73E-01

Sr-90	1.36E-01	7.07E-03	1.98E-02	7.46E-02	4.96E+00	9.19E-02	2.45E-01	9.32E-01
Ag-110m	1.99E+00	9.91E-02	2.80E-01	1.09E+00	1.76E+01	2.42E-01	7.93E-01	3.09E+00
Ag-111	3.75E+01	1.90E+00	5.32E+00	2.08E+01	1.45E+03	2.17E+01	6.82E+01	2.59E+02
Sb-125	1.09E-02	5.31E-04	1.52E-03	5.78E-03	4.36E-02	6.65E-04	1.90E-03	7.47E-03
Ru-103	6.59E-02	3.34E-03	9.76E-03	3.73E-02	1.86E+01	3.19E-01	8.85E-01	3.28E+00
Ce-144	7.90E-01	3.94E-02	1.14E-01	4.13E-01	1.27E+00	2.14E-02	6.13E-02	2.26E-01
La-140	6.07E-02	2.88E-03	8.54E-03	3.24E-02	7.30E-01	1.29E-02	3.54E-02	1.32E-01
Pu-239	1.58E-05	7.83E-07	2.22E-06	8.75E-06	2.41E-05	4.22E-07	1.16E-06	4.47E-06

8.2.1.2 Accident Scenario: Water Ingress

Table 8.10. 250 MW(t) pebble bed 700°C ROT—water ingress—short term attenuation factors.

Nuclide	Barriers to Fission Product Release (Attenuation Factors)			
	Accident – HPB		Accident – Reactor Building	
	DF 50%	DF 95%	DF 50%	DF 95%
Xe-133	1	1	1	1
Kr-85	1	1	1	1
Kr-88	1	1	1	1
I-131	2	1	10	2
I-133	2	1	10	2
Te-132	2	1	10	2
Cs-137	2	1	10	2
Cs-134	2	1	10	2
Sr-90	2	1	10	2
Ag-110m	2	1	10	2
Ag-111	2	1	10	2
Sb-125	2	1	10	2
Ru-103	2	1	10	2
Ce-144	2	1	10	2
La-140	2	1	10	2
Pu-239	2	1	10	2

Table 8.11. 250 MW(t) pebble bed 700°C ROT—water ingress—long term release fractions and attenuation factors.

Nuclide	Sources of Fission Product Release						Barriers to Fission Product Release (Attenuation Factors)											
	Fract. HM Contamination		Non-Intact (In-service failure)		Incremental Failure – Accidents		HM Accident Release		Non-intact Accident Release		Intact Accident Release		Accident – Graphite		Accident – HPB		Accident – Reactor Building	
	50%	95%	50%	95%	50%	95%	DF 50%	DF 95%	DF 50%	DF 95%	DF 50%	DF 95%	DF 50%	DF 95%	DF 50%	DF 95%	DF 50%	DF 95%
Xe-133	1E-05	2E-05	7E-06	2E-05	3E-05	2E-04	1	1	5	3.33	1E+07	2E+06	1	1	5	2	1	1
Kr-85	1E-05	2E-05	7E-06	2E-05	3E-05	2E-04	1	1	5	3.33	1E+07	2E+06	1	1	5	2	1	1
Kr-88	1E-05	2E-05	7E-06	2E-05	3E-05	2E-04	1	1	5	3.33	1E+07	2E+06	1	1	5	2	1	1
I-131	1E-05	2E-05	7E-06	2E-05	3E-05	2E-04	1	1	5	3.33	1E+07	2E+06	1	1	2	1	50	10
I-133	1E-05	2E-05	7E-06	2E-05	3E-05	2E-04	1	1	5	3.33	1E+07	2E+06	1	1	2	1	50	10
Te-132	1E-05	2E-05	7E-06	2E-05	3E-05	2E-04	1	1	5	3.33	1E+07	2E+06	1	1	2	1	50	10
Cs-137	1E-05	2E-05	1.7E-05	5E-05	3E-05	2E-04	1	1	1	1	1E+07	2E+06	5	2	2	1	50	10
Cs-134	1E-05	2E-05	1.7E-05	5E-05	3E-05	2E-04	1	1	1	1	1E+07	2E+06	5	2	2	1	50	10
Sr-90	1E-05	2E-05	1.7E-05	5E-05	3E-05	2E-04	1	1	1	1	1E+06	2E+05	50	10	2	1	50	10
Ag-110m	1E-05	2E-05	1.7E-05	5E-05	3E-05	2E-04	1	1	1	1	5E+02	1E+02	1	1	2	1	50	10

Ag-111	1E-05	2E-05	1.7E-05	5E-05	3E-05	2E-04	1	1	1	1	5E+02	1E+02	1	1	2	1	50	10
Sb-125	1E-05	2E-05	1.7E-05	5E-05	3E-05	2E-04	1	1	1	1	1E+06	2E+05	100	30	2	1	50	10
Ru-103	1E-05	2E-05	1.7E-05	5E-05	3E-05	2E-04	1	1	100	50	1E+07	2E+06	10	3	2	1	50	10
Ce-144	1E-05	2E-05	1.7E-05	5E-05	3E-05	2E-04	1	1	100	50	1E+07	2E+06	100	30	2	1	50	10
La-140	1E-05	2E-05	1.7E-05	5E-05	3E-05	2E-04	1	1	100	50	1E+07	2E+06	100	30	2	1	50	10
Pu-239	1E-05	2E-05	1.7E-05	5E-05	3E-05	2E-04	1	1	1000	500	1E+07	2E+06	1000	300	2	1	50	10

Table 8.12. 250 MW(t) pebble bed 700°C ROT—water ingress—source term (curies).

Nuclide	Release to Environment Short Term				Release to Environment Long Term			
	DBA	50%	Mean	95%	DBA	50%	Mean	95%
Xe-133	1.16E+01	1.16E+01	1.16E+01	1.16E+01	1.85E+02	4.58E+01	6.43E+01	1.85E+02
Kr-85	6.26E-02	6.26E-02	6.26E-02	6.26E-02	1.31E+00	3.08E-01	4.42E-01	1.31E+00
Kr-88	6.00E+00	6.00E+00	6.00E+00	6.00E+00	5.33E-04	1.26E-04	1.82E-04	5.33E-04
I-131	6.41E+00	3.21E-01	5.42E-01	1.80E+00	2.40E+02	1.33E+00	2.80E+00	1.02E+01
I-133	1.15E+01	5.76E-01	9.78E-01	3.27E+00	9.70E+01	5.32E-01	1.20E+00	4.41E+00
Te-132	8.80E+00	4.29E-01	7.51E-01	2.49E+00	2.39E+02	1.37E+00	2.95E+00	1.10E+01
Cs-137	1.63E+01	8.22E-01	1.40E+00	4.64E+00	2.84E+01	1.03E-01	2.89E-01	1.10E+00

Cs-134	2.62E+00	1.30E-01	2.26E-01	7.49E-01	3.12E+01	1.17E-01	3.15E-01	1.23E+00
Sr-90	2.78E+00	1.37E-01	2.31E-01	7.50E-01	5.30E+01	1.83E-01	5.38E-01	2.00E+00
Ag-110m	3.94E+01	1.95E+00	3.30E+00	1.11E+01	6.43E+01	2.45E-01	6.59E-01	2.53E+00
Ag-111	7.93E+02	3.95E+01	6.81E+01	2.29E+02	5.50E+03	2.11E+01	5.67E+01	2.12E+02
Sb-125	2.11E-01	1.06E-02	1.82E-02	5.99E-02	2.48E-01	8.10E-04	2.41E-03	9.48E-03
Ru-103	1.35E+00	6.72E-02	1.14E-01	3.77E-01	6.65E+01	3.17E-01	7.55E-01	2.79E+00
Ce-144	1.61E+01	8.16E-01	1.37E+00	4.56E+00	4.53E+00	2.14E-02	5.23E-02	1.95E-01
La-140	1.23E+00	6.09E-02	1.02E-01	3.44E-01	2.70E+00	1.24E-02	3.15E-02	1.16E-01
Pu-239	3.20E-04	1.58E-05	2.70E-05	8.93E-05	8.78E-05	4.12E-07	1.01E-06	3.77E-06

8.3 Radiation doses due to accidents in pebble bed HTGR

The doses at the exclusion area boundary will be presented for two cases. In the first, covering case A – conditions at exclusion area boundary and B – conditions at low population zone boundary – the meteorological conditions will be taken in accordance with US NRC guidance as the values of atmospheric dispersion coefficient covering 95% of possible atmospheric conditions.

In the second case shown as C, corresponding to calculation according to the requirements of PAGs, atmospheric conditions will be assumed to be mean values of atmospheric dispersion coefficient.

In the USA, the limiting doses for each of these cases are different, as discussed in chapter 7. In Poland, the dose limit after an accident without core melt is 10 mSv at exclusion area boundary. Since the aim of the work is to determine whether the pebble bed HTGR may be located in a small distance from an industrial enterprise, which will be using its energy, there is no need to calculate the values of possible doses to infants and children.

The proposed distance from the reactor center to exclusion area boundary is 400 m. The estimates presented above in chapter 4 for the radious of EAB = 425 m show that this distance is fully reasonable and should satisfy the requirements of safety both in the USA and in Poland.

*8.3.1 **Dose conversion factors***

The conversion from source terms (Ci) to rem shown below is done for the 95% meteorology

Table 8.13. Dose conversion factors—Cases A/B[52] 400 meters (rem/curie).

Nuclide	Short Term DCF 95% Meteorology	Long Term DCF 95% Meteorology
Xe-133	4.67E-07	3.17E-07
Kr-85	3.91E-08	2.62E-08
Kr-88	3.05E-05	1.01E-05
I-131	2.00E-03	1.14E-03
I-133	3.07E-04	1.54E-04
Te-132	7.46E-04	5.17E-04
Cs-134	3.02E-03	1.94E-03
Cs-137	2.02E-03	1.20E-03
Sr-90	1.02E-02	8.97E-03
Ag-110m	5.04E-03	3.21E-03
Ag-111	3.03E-04	2.18E-04
Sb-125	2.08E-04	1.23E-04
Ru-103	5.16E-04	3.65E-04
Ce-144	2.00E-02	1.29E-02
La-140	5.16E-04	2.72E-04
Pu-239	1.15E+01	1.10E+01

The conversion below is done for mean meteorology

[52] Cases A - Exclusion Area Boundary and B - Low Population Zone

Table 8.14. Dose conversion factors—Case C (rem/curie).

Nuclide	Short Term DC Mean Meteorology		Long Term DCF Mean Meteorology	
	400m	800m	400m	800m
Xe-133	1.87E-07	9.38E-08	1.90E-07	9.71E-08
Kr-85	1.43E-08	7.17E-09	1.57E-08	8.05E-09
Kr-88	1.36E-05	7.00E-06	4.43E-06	2.29E-06
I-131	6.18E-04	2.14E-04	6.92E-04	2.40E-04
I-133	1.19E-04	4.16E-05	9.23E-05	3.23E-05
Te-132	2.68E-04	9.50E-05	2.82E-04	1.01E-04
Cs-134	9.42E-04	3.28E-04	1.09E-03	3.81E-04
Cs-137	6.54E-04	2.28E-04	7.59E-04	2.64E-04
Sr-90	4.36E-03	1.51E-03	5.20E-03	1.80E-03
Ag-110m	1.64E-03	5.70E-04	1.90E-03	6.61E-04
Ag-111	1.13E-04	3.90E-05	1.26E-04	4.37E-05
Sb-125	6.65E-05	2.38E-05	7.27E-05	2.60E-05
Ru-103	1.93E-04	6.74E-05	2.19E-04	7.67E-05
Ce-144	6.81E-03	2.35E-03	8.06E-03	2.79E-03
La-140	1.64E-04	5.98E-05	1.49E-04	5.43E-05
Pu-239	5.61E+00	1.94E+00	6.65E+00	2.30E+00

8.3.2 Break in Helium Pressure Boundary of pebble bed HTGR

Table 8.15. 250 MW(t) pebble bed 700°C ROT—break in HPB—Cases A and B—400 m from release (rem).

Nuclide	Case A/EAB (400 m) /TEDE		Case B/LPZ (400 m)/ TEDE	
	Short	Long	Short	Long
Xe-133	5.48E-06		5.48E-06	2.24E-05
Kr-85	2.37E-09		2.37E-09	1.26E-08
Kr-88	1.86E-04		1.86E-04	2.00E-09
I-131	7.01E-04		7.01E-04	4.98E-02
I-133	1.72E-04		1.72E-04	2.74E-03
Te-132	3.44E-04		3.44E-04	2.30E-02
Cs-137	1.61E-03		1.61E-03	3.58E-03
Cs-134	3.70E-04		3.70E-04	6.37E-03
Sr-90	1.38E-03		1.38E-03	4.45E-02

Ag-110m	1.00E-02		1.00E-02	5.65E-02
Ag-111	1.14E-02		1.14E-02	3.15E-01
Sb-125	2.27E-06		2.27E-06	5.36E-06
Ru-103	3.40E-05		3.40E-05	6.78E-03
Ce-144	1.58E-02		1.58E-02	1.64E-02
La-140	3.13E-05		3.13E-05	1.99E-04
Pu-239	1.82E-04		1.82E-04	2.65E-04
Sum	**4.22E-02**	**0.00E+00**	**4.22E-02**	**5.25E-01**
TOTAL	**4.22E-02**		**5.67E-01**	
Limits (rem)	25 in USA , 1 in Poland			

The limit value in Poland is TEDE = 10 mSv, so the value of TEDE calculated for case B – that is for 30 days – equal to 0.567 rem = 5,67 mSv is below the limiting dose in Poland, but close. Since the dose limit in Poland is set for 12 months, before submitting the final calculations to Polish regulatory body it will be necessary to perform site specific calculations but this preliminary evaluation shows that the distance of about 400 m should be acceptable.

Table 8.16 250 MW(t) pebble bed 700°C ROT—break in HPB—Cases C and D—400 m from release (rem).

Nuclide	CASE C/Inh, grndshine, cldshine/TEDE		CASE D/Inh, grndshine, cldshine/Thyroid	
	Short	Long	Short	Long
Xe-133	2.19E-06	5.13E-06		
Kr-85	8.68E-10	2.89E-09		
Kr-88	8.30E-05	3.34E-10		
I-131	3.06E-05	1.79E-03	9.70E-04	5.72E-02
I-133	9.77E-06	9.61E-05	2.65E-04	2.60E-03
Te-132	1.75E-05	7.48E-04	2.82E-04	1.21E-02
Cs-137	7.73E-05	1.04E-04		
Cs-134	1.68E-05	1.66E-04		
Sr-90	8.65E-05	1.27E-03		

Ag-110m	4.59E-04	1.51E-03		
Ag-111	6.01E-04	8.60E-03		
Sb-125	1.01E-07	1.38E-07		
Ru-103	1.88E-06	1.94E-04		
Ce-144	7.75E-04	4.94E-04		
La-140	1.40E-06	5.27E-06		
Pu-239	1.25E-05	7.72E-06		
Sum	**2.17E-03**	**1.50E-02**	**1.52E-03**	**7.19E-02**
TOTAL	**1.72E-02**		**7.34E-02**	
Limits (rem)	1 in USA, 1 in Poland		5 in USA, 1 in Poland	

In the case of using mean meteorology the values of radiation doses at the distance of 400 m. are well below the limit dose values inPoland and in the USA.

8.3.3 Accident Scenario: Water Ingress Event

Table 8.17. 250 MW(t) pebble bed 700°C ROT—water ingress—Cases A and B—400 m from release (rem).

Nuclide	Case A/EAB (400 m)/TEDE		Case B/LPZ (400 m)/TEDE	
	Short	Long	Short	Long
Xe-133	5.42E-06		5.42E-06	5.86E-05
Kr-85	2.45E-09		2.45E-09	3.42E-08
Kr-88	1.83E-04		1.83E-04	5.38E-09
I-131	1.28E-02		1.28E-02	2.74E-01
I-133	3.54E-03		3.54E-03	1.49E-02
Te-132	6.56E-03		6.56E-03	1.24E-01
Cs-137	3.29E-02		3.29E-02	3.41E-02
Cs-134	7.92E-03		7.92E-03	6.05E-02

Sr-90	2.83E-02		2.83E-02	4.76E-01
Ag-110m	1.98E-01		1.98E-01	2.06E-01
Ag-111	2.40E-01		2.40E-01	1.20E+00
Sb-125	4.40E-05		4.40E-05	3.05E-05
Ru-103	6.98E-04		6.98E-04	2.43E-02
Ce-144	3.22E-01		3.22E-01	5.84E-02
La-140	6.36E-04		6.36E-04	7.33E-04
Pu-239	3.68E-03		3.68E-03	9.66E-04
Sum	**8.58E-01**	**0.00E+00**	**8.58E-01**	**2.47E+00**
TOTAL	**8.58E-01**		**3.33E+00**	
Limits (rem)	25 in USA, 1 in Poland			

Table 8.18. 250 MW(t) pebble bed 700°C ROT—water ingress—Cases C and D—400 m from release (rem).

Nuclide	CASE C/Inh, grndshine,		CASE D/Inh, grndshine,	
	Short	Long	Short	Long
Xe-133	2.17E-06	1.22E-05		
Kr-85	8.96E-10	6.94E-09		
Kr-88	8.16E-05	8.07E-10		
I-131	3.35E-04	1.94E-03	1.06E-02	6.19E-02
I-133	1.16E-04	1.11E-04	3.16E-03	3.00E-03
Te-132	2.01E-04	8.32E-04	3.24E-03	1.35E-02
Cs-137	9.13E-04	2.19E-04		
Cs-134	2.13E-04	3.44E-04		
Sr-90	1.01E-03	2.80E-03		
Ag-110m	5.42E-03	1.25E-03		
Ag-111	7.69E-03	7.15E-03		
Sb-125	1.21E-06	1.75E-07		

Ru-103	2.20E-05	1.65E-04		
Ce-144	9.34E-03	4.22E-04		
La-140	1.68E-05	4.69E-06		
Pu-239	1.51E-04	6.72E-06		
Sum	**2.55E-02**	**1.53E-02**	**1.70E-02**	**7.84E-02**
TOTAL	**4.08E-02**		**9.54E-02**	
Limits (rem)	1 in USA, 1 in Poland		5 in USA, 1 in Poland	

The results of table 8.17, case B (exposure for 30 days) show that the accident with water ingress results in radiation doses higher than the doses established as admissible in Poland (10 mSv = 1 rem for exposure during 1 year after the accident).

The difference is large so that the distance about 400 m would not be possible, unless the design of the reactor is modified so that the probability of water ingress accident can be reduced such that the accident is considered as beyond design basis accident.

Alternatively, the distance to exclusion area boundary can be increased or the local meterological conditions can be shown to be much better than those assumed in the calculations above.

*8.3.4 **Conclusions***

The review of pebble bed HTGR design has shown excellent safety features of this reactor. The evaluation of radiological consequences of various possible accidents has shown that there is only one accident which results in elevated radiation doses in the distance of 400 m from the reactor. That accident is water ingress to the primary circuit, and under 95% meteorology it leads to radiation doses over 1 month after the accident equal to 3,33 rem = 33,3 mSv TEDE.

This is the value of radiation dose which is fully acceptable in the USA and in several other countries in the European Union. However, in Poland the limiting dose at EAB is 10 mSv for a person staying at the boundary for 1 year and eating locally grown food products Therefore the danger of water ingress into the pebble bed HTGR has to be prevented. This means the necessity of design analysis from this standpoint. Other accidents do not involve hazards exceeding admissible values.